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SHIELDING: AN ANNOTATED BIBLIOGRAPHY

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FEBRUARY 1962

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SHIELDING:
AN ANNOTATED BIBLIOGRAPHY

Compiled by
WILLIAM L. HOLLISTER

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NOTICE

ABSTRACT

The following bibliography was compiled in connection with a study of radiation hazards upon the lunar surface. The bibliography includes works on the subject of reactor shielding both on land and in nuclear powered space vehicles, shielding against trapped radiation belts, cosmic rays, and solar flares. Both theoretical and empirical studies are included.

References were obtained from a search of the following sources (1958-date):

- LMSC Technical Information Center collection
- Environmental Effects on Materials and Equipment
- IAS Abstracts
- REIC Accessions List
- Nuclear Science Abstracts

Search Completed January 1962.

TABLE OF CONTENTS

	Page
Abstract	i
Table of Contents	ii
References	1
Corporate Author Index	147
Subject Index	152

1. Acraman, W. E.

GASLIT: A PROGRAMME FOR CALCULATING THE GAMMA DOSE RATE IN THE SHIELD OF A SMALL REACTOR. United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, Eng. Sep 60, 40p.

GASLIT (GAmma Shielding LIdo Type) is a program in MERCURY Autocode. It is designed to calculate the γ dose rate or the rate of heat deposition at points along the center line of a multilayer shield of a reactor due to sources in the core, at the sides of the core, and in the shield regions. "Taylor" build-up factors are used for the build-up of low energy photons. The source distribution may be expressed by a maximum of 700 values in the core, 500 at the sides of the core, and 144 mesh intervals in each of up to 30 shield regions. Lateral variation of the sources in each shield region may be allowed for by specifying the off axis source distribution with up to 40 equally spaced radial mesh intervals.

2. Agresta, J., Slater, M. and Soodak, H.

VALIDITY OF DIFFUSION THEORY FOR SHIELDING ANALYSIS. Nuclear Development Corp of America, White Plains, N. Y. Rept. no. NDA-2130-2, 31 Dec 59, 42p. (Contract AT(30-1)-2303, Proj. V)

The errors involved in the case of diffusion theory for handling one-velocity problems, as opposed to the more exact transport theory, were studied. Determinations of flux, given a distributed source, and of particle transmissions through finite regions were made. In each instance, transport theory solutions were obtained and compared with diffusion theory for various values of scattering-to-total cross section ratio and for various degrees of scattering anisotropy. Three types of problems were studied. An infinite medium with a source that varied sinusoidally with the single space coordinate y, a source which varied exponentially with the single space coordinate y in an infinite medium, and transmission of neutrons through finite slabs. Detailed discussions and tabulations were made for evaluation of diffusion theory.

3. Åkerhielm, F.

TRANSMISSION OF THERMAL NEUTRONS THROUGH BORAL. Aktiebolaget Atomenergi, Stockholm. AE-24, Aug 59, 17p.

Transmission measurements were performed using Maxwellian distributed neutrons from the R-1 reactor perpendicularly incident upon a boral absorption plate. American, English, German, Swedish, and Swiss samples were investigated and the results are compared to calculated values. The influence of the absorber grain size is discussed.

4. Åkerhielm, F.

TRANSMISSION OF THERMAL NEUTRONS THROUGH BORAL. Aktiebolaget Atomenergi, Stockholm. Rept. no. AE-24 (2nd Rev.) Aug 60, 20p.

Transmission measurements were performed using Maxwellian distributed neutrons from the Rl reactor perpendicularly incident upon a boral absorption plate. American, English, German, Swedish, and Swiss samples were investigated, and the results are compared to calculated values. The influence of the absorber grain size is discussed.

5. Allen, F., et al C

THE TRANSMISSION OF MONOENERGETIC NEUTRONS THROUGH BORATED POLY-ETHYLENE (U). Ballistic Research Labs., Aberdeen Proving Ground, Md. BRL rept. no. 1129; Proj. 512-10-001, Apr 61, 75p. ASTIA AD-324 077. CONFIDENTIAL REPORT

A description of a versatile Monte Carlo neutron transport code for multilayer, laterally infinite, plane shields is presented. Curves are given which show the principal results obtained for the transmission of monoenergetic neutrons through,

5. (cont'd) and their reflection from, polyethylene slabs containing 8 wt-% boron carbide. A discussion of the results is also given.

6. Allen, R. I., et al

SHIELDING PROBLEMS IN MANNED SPACE VEHICLES. Lockheed Nuclear Products, Marietta, Ga. Semiannual technical summary rept. 31 Dec 59 - 30 June 60; rept. no. NR-104, 124p. (Contract DA-01-009-506-ORD-832)

Results are presented from the first phase of a study of the shielding requirements for manned space vehicles, as imposed by the natural radiations to be encountered in space flight and the radiations from auxiliary nuclear power sources. The penetrating radiations in space are discussed and the reliability of various data is considered. Intensity, energy spectra, and spatial and time dependence of the various types of radiation are summarized in graphical and tabular form. The spatial distribution of trapped protons and electrons is presented in the form of contours of constant flux. The significance and general shielding implications of the more important penetrating radiations are treated. Computational methods for predicting dose rates due to trapped protons and electrons are formulated, and results of calculations are presented. Proton dose-rate calculations are based on the energy spectrum as determined by Freden and White, bremsstrahlung dose rates are calculated on the basis of the low-altitude electron spectrum measurements of Walt et al., and of Holly and Johnson. It is found that attenuation of protons in shields of moderate thickness is governed largely by ionization energy losses; whereas inelastic nuclear collisions result in attenuation of the primary flux of protons and in the production of secondary radiations, which become significant for very thick shields. Proton dose rates at the center of the proton belt range from 20 rads/hr behind a 2 g/cm^2 shield to about 1 rad/hr behind a 100 g/cm^2 carbon shield. Light elements are most effective for proton shields since they offer combined advantages of large stopping power and relatively small production of low-energy secondary radiation. Carbon is a particularly effective element for proton shielding; its stopping power is exceeded only by that of hydrogen and helium. Hydrogenous compounds may prove to be suitable since the stopping power of hydrogen is more than twice that of any other element; and such compounds offer advantages of attenuation of both protons and secondary neutrons by nuclear collisions. Production of bremsstrahlung by electrons stopped in the vehicle exterior can be minimized by use of a thin outer layer of low-Z material. Bremsstrahlung dose rates in a lightly shielded vehicle reach values of several r/hr but are reduced to acceptable levels by 3 or 4 g/cm^2 of lead or uranium.

7. Anderson, D. C. and Shure, K.
CALCULATION OF THERMAL NEUTRON FLUXES IN PRIMARY SHIELDS. Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh, Pa. Rept. no. WAPD-TM-193, Nov 59, 44p. (Contract AT-11-1-GEN-14)

A method is presented for calculating thermal neutron fluxes in the primary shields of reactor systems which eliminates reliance on mock-up experimental data. A multigroup P_1 approach is employed with the spatial dependence of the neutron attenuation adjusted through use of a point source attenuation kernel for a homogeneous hydrogeneous medium. Comparison of calculation with experiment is presented.

8. Anderson, D. C., Herwig, L. O., and Vogelsang, W. F.
COMPARISON OF MEASURED AND CALCULATED THERMAL NEUTRON DISTRIBUTIONS IN HIGH TEMPERATURE WATER SHIELDING. Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh, Pa. Rept. no. WAPD-T-1310, Mar 60, 13p. (Contract AT-11-1-GEN-14)

Measurements taken in the core and reflector of the High Temperature Test Facility at Bettis are presented. The measurements are compared with calculated thermal neutron distributions obtained by spatially integrating green's function, thus indirectly comparing the Green's function with experiment.

9. Anderson, K. A. and Fichtel, C. E.
DISCUSSIONS OF SOLAR PROTON EVENTS AND MANNED SPACE FLIGHT. National Aeronautics and Space Administration.

9. (cont'd) Rept. no. NASA-TN D-671, Mar 61.

The two papers which comprise this report deal with the prediction of solar beam events and the radiation hazard in space that results from them. These reports are in a preliminary form and further work is being done by both authors. However, owing to the continuing demand for information on this subject, and because these represent the most comprehensive surveys that have been carried out, they are being made available in their present form.

10. Asada, T., et al

Scattering of γ rays in the shield

labyrinth. Osaka University. J.

ATOMIC ENERGY SOC. JAPAN v. 2,

p. 245-52, May 1960. (In Japanese)

In a Hot Cave labyrinth system, the dose rate of scattered γ rays was difficult to calculate accurately. An approximate formula was derived by which the dose rate of scattered γ rays coming out of the labyrinth is obtained. The value calculated by the approximate formula was compared with the measured value of the actual Hot Cave as well as with that of models and found to be almost identical, the error staying within 10%. In this formula, the multiple scattering inside the shielding was ignored and the scattering was taken as being isotropic. These approximations were shown to be satisfactory at the labyrinth where it bends at right angles. By this formula the labyrinth shape which should be adopted for Hot Caves in order to obtain the most effective shielding was computed. The calculated value was compared with the actual value measured in both a lead and ordinary concrete.

11. Ashley, R. L.

Effective-energy method for spent-fuel

shielding. NUCLEONICS v. 16, n. 10,

p. 78-81, Oct 1958.

A curve computed from the yields of hard gamma-rays (<1.5 Mev.) emitting fission products is first used to determine the photon-emission power as a function of irradiation time and decay time. A second set of curves enables an effective photon energy (having the same attenuation in a particular shield as a mixture of emitted gamma-rays) to be determined. These curves are presented for lead thickness 6-12 inches, water 6-12 ft., and magnetite concrete 2-4 ft. The dose rate computed in this way is reasonably accurate for decay times <45 min.

12. Ashley, R. L.

GRAPHICAL AIDS IN THE CALCULATION OF THE SHIELDING REQUIREMENTS FOR SPENT U^{235} FUEL. Atomics International Div., North American Aviation, Inc., Canoga Park, Calif. Rept. no. NAA-SR-1992, 15 Nov 57, 59p. (Contract AT-11-1-GEN-8)

The shielding requirements for spent U^{235} fuel (or gross fission products) can be determined by using a method in which a single predetermined energy represents the entire fission product gamma-ray spectrum. The data presented herein, in the form of a series of graphs, can be used to obtain the value of this energy. Having this energy, the requisite shielding calculations can be performed using linear absorption coefficients and buildup factors which correspond to this energy. In essence, this method for determining fission product shielding requirements has as its basis a scheme termed the "effective energy" concept in which the entire fission product gamma-ray spectrum is replaced by a single gamma-ray line, behaving exactly as the entire spectrum for the given parameters of irradiation and decay time, shield material and thickness. The equivalence stems from the procedure utilized in determining the effective energy, i.e., the gamma rays comprising the fission product spectrum present at the time of interest are individually attenuated through a given shield and the over-all dose rate attenuation determined. The effective energy corresponds to the particular gamma ray which, if present in the same intensity as the fission product spectrum, would result in an equivalent dose rate when confronted by the same shield. The fission product spectrum considered consists, for the most part, of gamma rays with energies equal to or greater than 1.60 Mev. Since the range of shield thicknesses analyzed in the thick-shield category, it was unnecessary to consider gamma rays with lower energies as they would contribute negligibly to the dose rate at the shield surface. The range of exposure times considered is from 100 hours to 300 days, with decay times of from 20 min to 300 days. Monolithic shields of 6 to 12 in. of Pb, 5 to 12 ft. of H_2O, 3 to 6 ft. of ordinary concrete, and 2 to 4.25 ft. of magnetite concrete are analyzed. A comparison of the results obtained using the effective energy method with several experimental measurements indicates that this method yields data which are consistently lower than measured values. The ratio of calculated to measured values generally is greater than 0.5, but less than 0.9, with the greater deviation occuring for cooling times less than 45 min.

13. Ashley, R. L. and Duncan, D. S.

THE THEORETICAL CALCULATION OF THE ATTENUATION OF GAMMA RADIATION FROM A SWIMMING POOL TYPE REACTOR. North American Aviation, Inc., Atomics International Div., Canoga Park, Calif. Rept. no. NAA-SR-1921, 15 Aug 57, 40p. (Contract AT-11-1-GEN-8)

In order to determine the adequacy of the calculational methods employed by Atomics International to solve shielding problems on such reactor systems as the SRE and OMRE, it is required that the results obtained by using these methods be compared with accurate experimental information. Since the only reactor on which such experimental information is available is the Oak Ridge Bulk Shielding Reactor, the theoretical attenuation of gamma radiation in the BSR shield was evaluated. The results were compared with the measurements of the gamma-ray attenuation from this reactor out to a distance of 700 cm from the core. The analysis performed indicates that it is possible to reproduce the measured data quite accurately over the entire range of water thickness for which measurements have been made, the average value of the calculated to measured dose rates being 1.25 with the maximum deviation from this value being less than 19%. The formulation and assumptions necessary in the analysis are discussed. It is concluded that the spectra and methods used will yield results which are more than adequate when dealing with shielding analyses, per se and are probably adequate for heat generation problems as well.

14. Associated Lead Manufacturers Ltd.

AN IMPROVED SHIELDING MATERIAL FOR USE AGAINST INJURIOUS RADIATION. BRITISH PATENT 868,524, 17 May 61.

A description is given of a slab or other thick shaped piece of rigid material for use in the construction of an enclosure affording protection against radiation. The slab consists of unplasticized polyethylene or polypropylene with uniformly dispersed powdered lead or tungsten. The metal to plastic ratio is within the range 1:1 to 30:1. Boron is also added uniformly in an amount of 1% of the total weight. The slab is fabricated in a mold which allows air to escape but does not allow a substantial amount of the plastic material to escape. The mold has an unheated base plate and heated top ram and side walls. Space is left between the side walls and base plate to allow air to escape.

15. Glass against radiation. ATOMICS AND ENERGY v. 9, p. 202-3, 211, June 1958.

Special shielding glass windows are designed to give maximum shielding in minimum thickness, coupled with excellent visibility, resistance to darkening under the action of radiation, and low cost.

16. Auslender, S. and Trubey, D. K. INSTRUCTIONS FOR THE OPERATION OF AN ORACLE CODE FOR A MONTE CARLO SOLUTION OF THE TRANSPORT PROBLEM FOR GAMMA RAYS INCIDENT UPON A SLAB. Oak Ridge National Lab., Tenn. Rept. no. CF-60-10-37, 26 Oct 60, 39p.

A program has been coded for the ORACLE which will solve, using Monte Carlo techniques, the transport problem for monodirectional, monoenergetic γ radiation incident at angle θ upon an infinite laminated slab of finite thickness. The code is designed to give estimates of energy deposition, energy flux, tissue dose rate, reflected and transmitted energy current, and the angular and energy distribution of the reflected and transmitted energy current.

17. Axelrad, I. R. NEUTRON ABSORPTION AND SHIELDING DEVICE. U. S. Atomic Energy Commission. U. S. PATENT 2,942,116, 21 June 60.

A neutron absorption and shielding device is described which is adapted for mounting in a radiation shielding wall surrounding a radioactive area through which instrumentation leads and the like may safely pass without permitting γ or neutron radiation to pass to the exterior. The shielding device comprises a container having at least one nonrectilinear tube or passageway means extending there-through, which is adapted to contain instrumentation leads or the like, a layer of a substance capable of absorbing γ rays, and a solid resinous composition adapted to attenuate fast-moving neutrons and capture slow-moving or thermal neutrons.

18. Axelrad, I. R.

SETTLING NEUTRON RADIATION SHIELDING MATERIAL. U. S. Atomic Energy Commission. U. S. PATENT 2,961, 415, 22 Nov 60.

A settable, viscous, putty-like shielding composition is described. It consists of an intimate admixture of a major proportion of a compound having a ratio of hydrogen atoms to all other atoms therein within the range of from 0.5:1 to 2:1, from 0.5 to 10% by weight of boron, and a fluid resinous carrier. This composition when cured is adapted to attenuate fast moving neutrons and capture slow moving neutrons.

19. Babb, D. D., Keller, J. W., and McCray, E.

CURVE FITS OF GAMMA-RAY DIFFERENTIAL-ENERGY SPECTRA. Convair, Fort Worth, Texas. Rept. no. MR-N-251; NARF-59-36T, 1 Nov 59, 41p. (Contract AF33(600)-38946)

The results of the Nuclear Development Association (NDA) moments-method calculations of the penetration of gamma rays through various media from point-isotropic sources were interpolated and extrapolated to give differential energy spectra for a larger set of initial and final energies. For each of the new pairs of energies the data were fitted analytically and tabulated as a function of penetration distance and atomic number. The function used for the fit is:

$$\ln \frac{f(t)}{\mu_0 r} = a_1(\mu_0 r)^2 + a_2(\mu_0 r) + a_3 (\mu_0 r)Z + a_4 Z + a_5 Z^2 + a_6$$

where f(t) is the function tabulated by NDA. The range of Z was divided into three sections and the fit made independently for each section. Thus, 18 coefficients were required for each of 104 energy pairs.

20. Barbieri, L. J. and Lampert, S.

The interdependence of manned spacecraft design and radiation

20. (cont'd) shielding. Ford Motor Co., Newport Beach, Calif. AERO/SPACE ENG. v. 20, n. 4, p. 14-15, Apr 1961.

The shielding requirements for an orbit 500 km above the earth's surface were determined from data presently available. It is pointed out that shielding a space vehicle for human occupancy depends upon many factors such as orbit and length of time in that orbit. The allowable radiation exposure of the occupants and the weight of the vehicle must receive primary consideration. Available data are reviewed for range-energy relationships of all radiations known to be encountered in space. The shielding properties of various materials are reviewed, and the relative merits of Al and Pb are considered in detail. It is concluded that shielding against protons presents the main problem. Calculations were made of the average radiation path length for high-energy protons or heavy particles through an average man, assuming that the man has the stopping power of water. A value of 50 cm was obtained, and it is shown graphically that all protons of energy equal to or less than 300 Mev will be completely absorbed by the average man. It is concluded that if sufficient shielding is provided to absorb 52-Mev protons in those regions of a space vehicle, where the astronauts will spend most of the time in orbit, and provided the radiation flux is no higher than is believed at present, there is no reason why a 4-week flight is not possible from a radiation hazard viewpoint. It is pointed out that solar flares remain the greatest unknown. The need is stressed for a handbook showing the attenuation, degradation, and transmission of the several types of radiation through a variety of structural materials as functions of thickness, and for the breakdown of the number of particles per unit energy interval as a function of energy.

21. Barbieri, L. J., Lampert, S. RADIATION SHIELDING AND MANNED SATELLITE DESIGN CONSIDERATIONS. Aeronutronic, Newport Beach, Calif. (Presented at the National IAS/ARS Joint Meeting, Los Angeles, Calif., 13-16 June 1961) (American Rocket Society, Inc., New York, N. Y., 61-163-1857)

Limits on mission duration predicted on the shielding afforded by various structural components are established. The extent to which additional shielding techniques may be relied on to increase the residence time in orbit is indicated. Examples are chosen to demonstrate the compatibility of radiation shielding offered by specific structural design features for a vehicle operating in a 300 mi. near-Earth orbit.

22. Barnes, T. G., Finkelman, E. M. and Barazotti, A. L.

Radiation shielding of lunar spacecraft.

(Amer. Astronautical Soc., Lunar Flight

Symposium, New York, N. Y. 27 Dec 1960)

ASTRONAUTICAL SCI. REV. v. 3, p. 11-18,

Jan -Mar 1961.

Review of current knowledge of the particle radiation which exists above the earth's atmosphere and which can be both biologically hazardous and damaging to sensitive spacecraft equipment. Implications for unmanned and manned cislunar flight are discussed from the point of view of the spacecraft designer. Consideration is given to galactic cosmic rays, solar protons, and both protons and electrons trapped in the earth's magnetic field. The main source of radiation damage, it is believed, is solar protons. Approaches to protecting equipment from this damage are noted, and shielding requirements for men are discussed. A possible corridor of flight paths which avoids the region of maximum radiation intensity is considered, and solar flare encounter probability is studied.

23. Batter, J. F., Jr. and Clarke, E. T.

MODELING AS A TECHNIQUE FOR

DETERMINING RADIATION SHIELDING.

Technical Operations, Inc., Burlington,

Mass.

The application of the laws of scaling to shielding experimentation provides an attractive, economical alternative to the testing of large structures, and permits the evaluation of shielding effectiveness of new structures while they are still in the blueprint stage. In theory, perfect scaling can be achieved if all dimensions are reduced and all densities of materials (including the atmosphere and the ground) are increased by the same scaling factor. In practice, an achievement of a scaling factor of 1:12, large enough to be useful, requires compromises. Air and ground, must in general, be left unchanged and building walls constructed of a substitute material of higher density with approximately the same attenuation cross sections, to prevent them from becoming unduly thick. The resultant model will then approximate the interaction of radiation with the structure but will ignore the effects of ground and air scatter. Equipment used in the preliminary experiments to test the utility of this concept and comparison of the results of the experiments with full-scale tests are described.

24. Beaugé, R.

Determination of the efficiency of a boron contained in this shield. (Centre d études Nucléaires, Fontenay-aux-Roses, France). INDS. ATOMIQUES v. 4, n. 3-4, p. 65-7, 1960. (In French)

The neutrons coming from a reactor channel are not monochromatic, and their absorption does not follow a simple exponential law. The attenuation law for thermal neutrons counted by a bor on counter is determined as a function of the thickness of an absorber in 1/v.

25. Beauge, R., Millot, J. P. and Rastoin, J.

ETUDE EXPÉRIMENTALE DE L'ATTENUATION DES NEUTRONS DANS LE BÉTON ORDINAIRE. (Experiments on neutron attenuation by ordinary concrete). Commissariat à l'Energie Atomique, Paris, France. Rept. no. CEA-1186, Apr 1959, 16p.

The penetration of fission neutrons in water and glass mixtures has been investigated in the NAIADE facility. The slopes of the thermal and fast fluxes (the latter measured with a dosimeter) remain similar when the volume proportion of water is greater than 15 per cent. For smaller water contents the measurements show the evidence of streaming, presumably due to slowing-down neutrons of energy smaller than 300 kev. Since a volume-proportion of about 15 per cent of water in glass corresponds to the composition of ordinary concrete, the present work enables prediction of the effects of a dessication of concrete to be made. Moreover, it is seen that there is no interest in increasing the proportion of water in ordinary concrete beyond the usual values (15 to 20 per cent).

26. Beauge, R., Millot, J. P. and Rastoin, J.

ETUDE EXPERIMENTALE DE L'ATTENUATION DES NEUTRONS DANS LES MELANGES PLOMB-EAU. (Experiments on neutron attenuation by water-lead mixtures) Commissariat à l'Energie Atomique, Paris, France. Rept. no. CEA-1187, May 59, 20p.

The penetration of fission neutrons in water-lead mixtures has been investigated in the NAIADE facility. The slopes of the thermal and fast fluxes (the latter measured with a dosimeter) remain similar when the volume proportion of water is greater than 65 per cent. For smaller water contents, the measurements show the evidence of streaming presumably due to slowing-down neutrons of energy smaller than 300 kev.

27. Beique, R. A. and Lougheed, M. N.

Considerations of shielding for Cesium-137 sources, containing Cesium 134. Ontario Cancer Inst., Toronto. RADIOLOGY v. 76, p. 281-6, Feb 1961.

Studies were made of shielding requirements for a 2000-c Cs^{137} source with a maximum additional activity of 5% Cs^{134}. It is pointed out that planning the shielding of a treatment head loaded with Co^{60} pure Cs^{137} involves calculation of the necessary number of half-value layers. The addition of Cs^{134} makes the problem more complex, since the higher three energies of Cs^{134} rely predominantly on density shielding factors rather than the photoelectric. Data are tabulated on the γ spectra of Cs^{134} and Cs^{137}. Graphs indicate the relative importance of photoelectric and Compton effects with relation to atomic number and energy, the transmission of γ radiation from Cs^{134}, Cs^{137}, and Co^{60} through concrete and lead, and the increase in output due to the addition of specific percentages of Cs^{134} to Cs^{137} used in γ sources.

28. Bendall, D. E.

A COMPUTER PROGRAMME FOR DETERMINING THE DISTRIBUTION OF PENETRA-

28. (cont'd) TING NEUTRONS IN REACTOR SHIELDS.

United Kingdom Atomic Energy Authority.

Research Group. Atomic Energy Research

Establishment, Harwell, Berks, England.

Rept. no. AERE-R-3082, Oct 59, 19p.

A program for use with the Ferranti mercury computer is described which evaluates the initial input source distribution data for the neutron shielding program RASH B. Alternatively, the program can be used to obtain the removal flux distribution through a reactor shield and core. The distribution of the average energy of the calculated quantity is also obtainable if required. The application of this program is limited to large cylindrical or plane systems.

29. Bendall, D. E.

A PROGRAMME FOR CALCULATING THE

GAMMA RAY FLUX THROUGH A MULTI-

LAYER SHIELD. United Kingdom Atomic

Energy Authority. Research Group. Atomic

Energy Research Establishment, Harwell,

Berks, England. Rept. no. AERE-R-2882,

Mar 59, 34p.

A program in autocode for the Ferranti Mercury Computer is described. It is designed to calculate the gamma ray flux due to a distributed source in a multi-layered shield in plane geometry. Up to 43 different shield regions may be included and each of these may contain up to a maximum of 144 mesh intervals. Build up of low energy gamma photons is treated by means of an average build-up factor expressed in the Taylor form. The output may be chosen to be gamma dose rate or rate of energy deposition.

30. Berga, J. O.

THE SPLIT SCATTER SHIELD FOR

SPACE APPLICATIONS. Air Force

Inst. of Tech., Wright-Patterson Air

30. (cont'd) Force Base, Ohio. (Master's thesis)

Rept. no. GNE/Phys/61-2, Mar 61, 54p.

The split scatter shield consists of multiple shield sections which are separated from each other. When this type of shield is used outside the atmosphere, radiation scattering in each section has a high probability of escaping the shield system without rescattering. Consequently, this shield has been proposed as a replacement for the normal shield unit construction for space applications. The split shield is compared with a unit shield of the same weight and material in a hypothetical problem situation by computing the radiation attenuation by each. The problem is computed using point and circular source geometries assuming the sources to emit gamma rays only. The calculations are performed using simplifying assumptions so that computer programs are not necessary. A geometrical parameter study is performed for the split shield to determine the optimum number of sections, the optimum separation distance between sections, and the maximum flux attenuation. The results indicate that a split shield is capable of greater flux attenuation by a factor of more than four than a unit shield of the same weight.

31. Blizard, E.P.

EXPERIMENTAL REACTOR SHIELDING

RESEARCH AT ORNL. Oak Ridge National

Lab., Tennessee.

The central shielding problem at ORNL is that of shielding a crew of a nuclear powered airplane. In addition to shielding experiments which are directly useable in shield design, research is directed to certain aspects of the basic shielding problem. A discussion is given of measurements of low energy distributions of neutrons in angle and energy. A lithium-iodide scintillation spectrometer developed for use in shielding measurements is described. Discussions are given of applications of the Lid Tank Shielding Facility and Tower Shielding Facility in shielding research.

32. Blizard, E. P.

ON THE DISK TO POINT OR INFINITE

PLANE SHIELDING TRANSFORMATIONS.

Oak Ridge National Lab., Tenn. Rept. no.

ORNL-2882, 5 Jan 59, 12p. (Contract

W-7405-eng-26)

32. (cont'd) The detector readings to be expected from an infinite plane isotropic source in a uniform infinite medium are derived in terms of measurements at specific distances along the axis of a uniform isotropic disk source. Extrapolation of the higher terms in this series on the basis of exponential attenuation is described, and approximations are given for the remainder in this series. Another transformation is given which yields directly the point-to-point attenuation kernel in terms of a series of measurements on the axis of a plane isotropic disk source. Approximations based on the extrapolation of the data are also given to enable estimation of the remainder of the series.

33. Blizard, E. P. and MacKellar, A. D.

Prediction of thermal-neutron fluxes in the BSF from LTSF data. Oak Ridge National Lab., Tenn. In Paper 6 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 6p.

Predictions of the thermal-neutron fluxes to be expected near the Bulk Shielding Reactor were made on the basis of Lid Tank Shielding Facility experimental data transformed to the BSR geometry. The predicted fluxes are higher than the measured fluxes by 15 to 25% for distances closer than 60 cm, but they are essentially in agreement for larger distances. At a distance of 125 cm, the predicted flux was only 1% too low.

34. Blizard, E. P. and Miller, J. M.

RADIATION ATTENUATION CHARACTERISTICS OF STRUCTURAL CONCRETE. Oak Ridge National Lab., Tenn. Rept. no. ORNL-2193, 29 Aug 58, 53p. (Contract W-7405-eng-26)

An expanded version of data previously published in ORNL-2081, p. 157, 1956. An experiment was performed at the Lid Tank Shielding Facility to determine the dependence of the neutron attenuation of a structural concrete shield on the amount and placement of water hydrogen throughout the shield. It was found that a 7 wt. % water content is adequate to ensure that intermediate-energy neutrons are quickly slowed

34. (cont'd) down to thermal energy, at which energy they are easily captured. A greater water content improves the over-all neutron attenuation according to the removal cross-section theory. In addition to the investigation of the neutron attenuation, the measured gamma-ray attenuation obtained in a calculation using buildup factors determined by the moments method. The two values were in good agreement.

35. Blizard, E. P.

SHIELD OPTIMIZATION. Oak Ridge National Lab., Tenn. Rept. no. ORNL-1471, 17 Mar 53, 20p. Decl. 2 Sep 60. (Contract W-7405-eng-26)

A theoretical treatment of shield optimization in nuclear-powered aircraft is given. An expression is developed for the ratio of neutron to γ dose rates in an optimized spherically symmetric shield. The general method for shield optimization in which a finite number of parameters is adequate is outlined, and the optimization of a box-shaped shield as an example of this method is considered.

36. Blosser, T. V., et al

A STUDY OF THE NUCLEAR AND PHYSICAL PROPERTIES OF THE ORNL GRAPHITE REACTOR SHIELD. Oak Ridge National Lab., Tenn. ORNL rept. no. 2195, 8 Sep 58, 43p. (Contract W-7405-eng-26)

A study of the nuclear and physical properties of the concrete shield of the ORNL Graphite Reactor was performed both to determine the radiation attenuation characteristics of the shield and to discover any effects of a long-term (12-year) irradiation. In the experiment neutron and gamma radiation measurements were made in a 4 5/8-in.-diam. hole in the shield as the hole was drilled in increments of 1 or 1/2 ft. Concrete plugs were always placed behind the detectors to simulate a homogeneous shield. The average relaxation length for gamma rays in the barytes-haydite concrete, of which most of the shield is made, was found to be approximately 13.6 cm, increasing gradually from 13.0 cm at a shield thickness of 2 ft. to 14.6 cm at a shield thickness of 5 ft. The fast-neutron relaxation length in the barytes-haydite concrete varied from 10.0 cm to 10.6 cm, the average being approximately 10.2 cm. Measurements of the streaming of radiation through the hole were also made. The concrete dust

36. (cont'd) collected from the drillings was used to determine the chemical composition, water content, density, compressive strength, and radioactivity of the shield at the various depths. The temperature gradient through the shield was also measured. This investigation showed that the chemical properties and density of the shield have not changed appreciably since a similar investigation in 1948, but its compressive strength is lower (~40 near the reflector-shield interface.

37. Block, E. and Jarnholt, M.

NUCLEAR REACTOR SHIELDING. PART I.

A LITERATURE SURVEY. (Aktiebolaget

Atomenergi, Stockholm). Rept. VDIT-32(1),

Mar 61, 45p.

Prepared for the EAES-Symposium on Nuclear Reactor Shielding, held at Studsvik, March 27-29, 1961. A literature survey including 171 abstracts is presented on both theoretical and experimental aspects of reactor shielding and covers research results published in two years up to March 1961. An author index is included.

38. Booker, D. V. and Flew, E. M.

ABSORPTION IN LEAD OF THE GAMMA

RADIATION FROM IRRADIATED URANIUM

SLUGS. Great Britain, Atomic Energy

Research Establishment, Harwell, Berks,

England. Rept. no. AERE-HP/R-1001,

15 Aug 52, 19p.

The absorption of the gamma radiation from two uranium slugs irradiated in BEPO for 781 days at an average power in the slug of 132 watts and for 20.5 days at an average power in the slug of 385 watts, respectively, was measured in lead. The results are given in three series of curves, showing gamma dosage against cooling time for different thicknesses of lead shield, fraction of gamma dosage transmitted, i.e., I/I_0, against thickness of lead shield required to give absorption factors of 10, 100, and 1000 against cooling time as deduced from the experimental results. Graphs are also plotted of the calculated curies of fission products in the energy ranges 0 to 0.5, 0.5 to 1.0, 1.0 to 1.6, and > 1.6 Mev, and the average energy for different cooling times. A table is given of the calculated average self absorption in the slugs at various cooling times.

39. Bowman, L.A. and Trubey, D.K.

STRATIFIED SLAB GAMMA-RAY DOSE-RATE BUILDUP FACTORS FOR LEAD AND WATER SHIELDS. Oak Ridge National Lab., Tenn. Rept. no. CF-58-1-41, 16 Jan 58, 10p. (Contract W-7405-eng-26)

The ORACLE Monte Carlo code for the calculation of the penetration of gamma rays through stratified slabs was used to calculate a total of 512 problems for eight different lead and water configurations. The energy of the incident radiation, the angle of incidence, the thickness of the shield, and the percentage of lead preceding or following water were varied. The source was assumed to be a monodirectional beam with energies of 1, 3, 6, and 10 Mev. The incident angles chosen were those which would give slant thicknesses of 1, 2, 3, and 4 times the normal thickness. The infinite slabs had finite normal thicknesses of 1, 2, 4, and 6 mean free paths. The results obtained included the dose rate and energy flux throughout the slab and at the rear; dose-rate buildup factors; the heat deposited throughout the slab; and the energy and angular distribution reflected and transmitted through the slab. Only the buildup factors at the rear of the slabs for normal incidence are considered in this report.

40. Braestrup, C.B. and Wyckoff, H.O.

RADIATION PROTECTION. Springfield, Ill., Charles C. Thomas, 1958, 374p.

Potential hazards for man resulting from developments in nuclear energy are reviewed. A summary is presented of the present knowledge of radiation protection. The material is presented in a manner understandable to professional people who are not specialists in the field. The material is applicable to the training of radiological defense groups.

41. Braley, J.E. and Donaldson, R.E.

NEUTRON PENETRATION BY THE MONTE CARLO METHOD. Convair, San Diego, Calif. Rept. no. ZJ-015, 29 May 57, 146p.

A calculation was made of the penetration of plane slabs of Pb and B loaded polyethylene by a plane parallel source of fast neutrons. The neutron energies used were

41. (cont'd) 14, 10, 6.5, and 3 Mev. The slab thickness ranged up to 6 mean free paths and in one instance to 9 mean free paths. The spectral distributions of the transmitted and reflected neutrons were recorded as well as the volume distributed source of γ rays resulting from captures and inelastic scatterings.

42. Breitling, G.

PRINCIPLES OF PERSONNEL RADIATION SHIELDING. (Universität, Tübingen, Germany). RÖNTGEN BL. v. 14, p. 65-72, Mar 1961. (In German)

On the basis of the intensity distribution of Compton's scattered radiation, the possibilities of danger offered by scattered rays of various radiation qualities are indicated. As an example, the distribution of dose on the diagnostic apparatus is presented, and measures for the reduction of the load are discussed. The dependence of the scatter and back-scatter on the tube voltage and the atomic number of the scattering material is described. In conclusion, the suitability of various building materials for shielding purposes is briefly mentioned.

43. Broder, D. L., et al

The effect of layers with boron content on the secondary gamma-ray yield. ATOMNAYA ENERG. v. 8, p. 49-51, Jan 1960. (In Russian)

The influence of a boron carbide layer, placed between steel and Plexiglas (imitating water), on the intensity of γ produced was studied in two types of steel, St-3 and stainless steel 1X18H9T. The geometry of the experiment, the spatial distribution of neutrons, and the γ spectrum formed are plotted.

44. Broder, D. L., et al

The passage of fast neutrons through lead and iron. ATOMNAYA ENERG. v. 7, p. 313-20, Oct 1959. (In Russian)

44. (cont'd) Results are presented from measurements made of the spatial distribution of fast neutrons in lead and iron from monoenergetic neutron sources (E_o = 4 and 14.9 Mev) and from fission. A kinetic equation for neutron attenuating media is used for calculating the spatial energy distributions of fast neutrons at large distances from the source. The anisotropy of elastic scattering is evaluated. The neutron energy losses in elastic scattering are neglected.

45. Brown, W. W. and Engholm, B. A.

MEASUREMENT OF NEUTRON ATTENUATION IN SHIELD MATERIALS. North American Aviation, Inc., Downey, Calif. Rept. no. NAA-SR-Memo-1143, 4 Nov 54, 39p. Decl. 14 Mar 57.

Measurement of the attenuation by iron, magnelite concrete, ordinary concrete, and aluminum, of the neutrons emerging from the top of the water boiler reactor thermal column is described. Attenuation of the thermal and epithermal neutrons was measured with indium foils, and that of neutrons above 1 ev, 0.5 Mev, and 1.5 Mev with a BF_3 proportional counter in paraffin moderator and Np^{237} and U^{238} fission counters respectively.

46. Broyles, A. A.

THE EFFECT OF SHELTERING ON FALLOUT FATALITIES. Florida Univ., Gainsville. Rept. NP-10038, p. 127-37.

The discussion includes: the determination of the infinite plane radiation dose probability; the biological response functions for surface fall-out gamma irradiation, and Sr^{90}; and the expected fraction of the population killed by a bomb. An estimate of the effects of world-wide fall-out indicates that something like 250,000 megatons would have to be exploded before 1% of the world's children are likely to die eventually from the effects of Sr^{90}.

47. Bunch, W. L.

ATTENUATION PROPERTIES OF HIGH DENSITY PORTLAND CEMENT CONCRETES

47. (cont'd) AS A FUNCTION OF TEMPERATURE.
General Electric Co. Hanford Atomic
Products Operation, Richland, Wash.
Rept. no. HW-54656, 22 Jan 58, 55p.
(Contract W-31-109-Eng-52)

The elevated temperatures used in the test appreciably impared the neutron and gamma attenuation properties of the two high density concretes. Nonetheless, the concretes would serve as satisfactory shield material if the reduced attenuation efficiency at the elevated temperatures were taken into account in the original reactor design. The two concretes used in the test were: (1) iron limonite, which consists of an aggregate of iron punchings and a grout of limonite, and iron ore containing water of hydration, and (2) magnetite limonite in which the non-hydrated iron ore magnetite is used as an aggregate and the limonite again is the grout material. Significant changes in attenuation effectiveness were found as the concretes were heated to 100 and 175°C, but very little change was effected in the final heating to 320°C. The density of the concretes was similarly affected, indicating that the bulk of the water had been evolved at 175°C and that the water which remained was more tightly bound. Because of this apparent saturation effect it should be possible to extrapolate the attenuation properties to slightly higher temperatures than those used if future laboratory tests were to show that no significant change in density and water content took place at the proposed temperature.

48. Burns, L. S., Jr.
Relative contributions of scattered and
secondary radiation inside a realistic
crew shield. General Electric Co.
Aircraft Nuclear Propulsion Dept.
Evendale, Ohio. In Paper 19 of
PAPERS FROM SEVENTH SEMI-ANNUAL
SHIELDING INFORMATION MEETING,
14-15 OCTOBER, 1959. TID-6302, 13p.

Analytical calculations were made of the dose rate inside a crew shield as a function of the emitted radiation energy and the angle of departure from the reactor shield. Scattering calculations were made by single scattering methods. Calculations were made for (n, α) capture in air, inelastic scattering in air, thermal neutron capture in the crew shield, and thermalization of fast neutrons and capture of those neutrons

48. (cont'd) in the crew shield. It is believed that this kind of study will prove highly useful in early planning stages of reactor shield design.

49. Burns, N. M. and Hanna, T. D.

AN INVESTIGATION INTO THE PHYSIOLOGICAL LIMITATIONS ASSOCIATED WITH THE USE OF ANTI-EXPOSURE SUITS - THE EFFECTS OF PROTECTIVE AVIATOR CLOTHING ON OPERATOR PERFORMANCE. Naval Air Material Center. Rept. no. NAMC-ACEL-463, 8 Sep 61.

50. Burrell, M. P. and Cribbs, D. L.

A Monte Carlo calculation of the transport of neutrons through iron slabs. Lockheed Aircraft Corp., Marietta, Ga. In Paper 13 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDINGS INFORMATION MEETING, 14-15 OCTOBER, 1959, TID-6302, 10p.

A Monte Carlo sampling is used to calculate the transmitted number current, number flux, dose, and the dose albedos of neutrons incident on an iron slab. A plane parallel source of monoenergetic neutrons with a number current density of 1 neutron/cm^2-sec is incident from a vacuum upon an iron slab of finite thickness. Transmission factors and linear buildup factors are computed for the above quantities. The dose transmission and the dose albedo are calculated for each of 120 equal solid angles. Gamma rays from inelastic scattering are recorded as a function of energy and penetration depth.

51. Burrus, W. R.

RADIATION TRANSMISSION THROUGH BORAL AND SIMILAR HETEROGENEOUS

51. (cont'd) MATERIALS CONSISTING OF RANDOMLY DISTRIBUTED ABSORBING CHUNKS. Oak Ridge National Lab., Tenn. Rept. no. ORNL-2528, 18 Jan 60, 30p. (Contract W-7405 eng-26)

Shields that consist of randomly distributed absorbing chunks in a relatively transparent matrix must contain a greater mass of absorber than homogeneous shields which provide the same attenuation. This is the result of radiation "channeling" between the absorbing chunks. Channeling is particularly important for heterogeneous materials when the mean free path for absorption is comparable to the chunk size. A newly developed method for calculating the transmission of radiation through such heterogeneous shields is described. The numerical results of a calculation of the transmission of thermal neutrons by boral (a B_4C-Al mixture) are given, including the effects of energy and angular distributions on the predicted attenuation. The calculated results are in reasonable agreement with available experimental results.

52. Campagna, E. R.

A DETERMINATION OF HALF-THICKNESS OF ZINC BROMIDE SOLUTION. Utah Univ., Salt Lake City. Rept. no. UUT-3, 22 July 53, 12p. (Contract DA-18-108-CML-4753)

The half-thickness of the concentrated solution of zinc bromide used as shielding material in the windows of the filling cell in Able Area at Dugway Proving Ground was investigated. The average energy of the gamma source used was 1.17 Mev, emitting a collimated beam through beakers containing the zinc bromide in 1/8 inch fluid increments from 0 to 1 1/2 inches to a Beckman ionization chamber Model MX-4. With all sources of error considered, the half-thickness was determined as 1.85 ± 0.02 inches.

53. Carsten, A. L. and Noonan, T. R.

Recovery from partial-body x-irradiation as measured by the lethality of two exposures. Univ. of Rochester, N. Y. RADIATION RESEARCH v. 11, p. 165-76, Aug 1959.

53. (cont'd) For the purpose of studying recovery from partial-body irradiation, the efficiencies of various shielding procedures have been evaluated. A double wraparound shield, which was the most efficient, reduced the rate to about 1% of the outside dose at a distance of approximately 6 cm from the edge of the shield. These studies, together with appropriate dissections, have permitted estimates of dosages to various organs under the conditions of exposure employed. In confirmation of previous work, the LD_{50}'s for upper-body, lower-body, and whole-body irradiation were found to be 1750 r, 1080 r, and 750 r, respectively. By using two partial-body exposures to x radiation, separated by various time intervals, the pattern of recovery (as measured by lethal effects) has been determined for lower- and upper-body irradiation. The recovery from a lower-body exposure of 648 r can be expressed by a single exponential function with a half-recovery time of 1.7 days. The recovery from an upper-body exposure of 1300 r can be expressed by a single exponential with a half-recovery time of 7.0 days or by two exponentials, one with a half-recovery time of 3.8 days and the second with a half-recovery time of 7.5 days.

54. Casper, A. W.

Comparison of calculated and measured water centerline data. General Electric Co. Aircraft Nuclear Propulsion Dept., Evendale, Ohio. In Paper 20 of PAPERS FROM SEVENTH SEMIANNUAL SCHIELDING INFORMATION MEETING, 14-15 OCTOBER 1 959. TID-6302, 9p.

Dose rates and thermal fluxes were measured in water at the ORNL Bulk Shielding Facility and at the GE Source Plate Facility at Battelle Memorial Institute. The measured values are compared with point kernel calculations of the dose rates and therm al fluxes calculated by a combined diffusion theory-point kernel approach.

55. Chapman, G. T. and Flynn, J. D.

A COMPARISON OF THE SHIELDING PROPERTIES OF IRON AND STAINLESS STEEL IN THE ORNL LID TANK (EXPT. 38) Oak Ridge National Lab., Tenn. Rept. no. CF-53-6-187, 30 June 53, Decl. 22 Apr 57, 29p.

55. (cont'd) (Contract W-7405-eng-26)

Thermal neutron flux, fast neutron dose, and λ dose measurements were made behind several Be-stainless steel and Be-Fe configurations. Also measurements were taken behind various thicknesses of stainless steel alone.

56. Chapman, G. T. and Flynn, J. D.

THERMAL NEUTRON FLUX BEYOND THE BOTTOM SHIELD MOCKUP. Oak Ridge National Lab., Tenn. Rept. no. CF-52-11-136, 5 Nov 52. Decl. with deletions 19 Apr 57, 8p.

The tested mockup was composed of Be, Fe, Pb, and H_2O. Neutron doses at various positions beyond the shield are plotted.

57. Chilton, A. B., Holoviak, D. and Donovan, L. K.

DETERMINATION OF PARAMETERS OF AN EMPIRICAL FUNCTION FOR BUILD-UP FACTORS FOR VARIOUS PHOTON ENERGIES. National Civil Engineering Lab., Port Hueneme, Calif. Technical note N-389. Interim report. Project Y-F011-05-402, Aug 60, 14p.

The use of a simple analytical expression for dose buildup factors from a radioactive isotropic point γ source is described. The parameters of the expression were determined by the method of least squares to obtain an optimum fit to experimental data for build-up factors for aluminum at various photon energies.

59. Clifford, C. E.

Gamma dose in a hole in a uniformly contaminated plane: Contribution by ground penetration. Defense Research Chemical

59. (cont'd) Labs., Ottawa, DRCL-310A. CAN. J. PHYS. v. 39, p. 604-8, Apr 1961.

A cylindrical hole in a uniformly γ-contaminated plane (e.g., a foxhole in a level field) is considered. The γ dose from ground penetration is measured as a function of depth in the hole, using a Cs^{137} source. Skyshine effects are not measured.

60. Clifford, C. E.

NEUTRON DOSE DISTRIBUTION BEYOND VARIOUS STRUCTURAL MEMBER MOCK-UPS. Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-1, Pt. II, 4 Oct 52. Decl. with deletions 19 May 57, 32p. (Contract W-7405-eng-26)

A series of experiments to determine the magnitude of neutron leakage from the steel structures which penetrate the top plug shield was done in the Lid Tank on the final part of the shield testing program. A series of mockups was constructed and installed in the Lid Tank. The mockups were built to simulate those features which were believed to represent the worst cases of streaming in the top plug design. The neutron traverses indicate roughly the size of a patch which would be required to adequately shield such weak spots in the shield.

61. Clifford, C. E.

NEUTRON AND GAMMA DOSE DISTRIBUTION BEHOND BERYLLIUM-SOLID IRON SHIELD IN WATER. Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-1, Pt. 2, 14 Aug 52. Decl. with deletions 19 Apr 57, 20p. (Contract W-7405-eng-26)

The beryllium-solid iron-water measurements were taken to aid in determining the feasibility of eliminating the boron carbide in the high-temperature regions of primary and top plug shields. A solid iron layer beyond the beryllium would become less effective as a gamma shield with increasing thickness because of the excessive produc-

61. (cont'd) tion of capture gammas in and beyond the slab. This would tend to increase the over-all shield weight somewhat over that for the boron carbide shield. Measurements were taken of both the gamma and neutron dose distribution beyond the iron in the beryllium, iron, water configuration. The measurements of the gamma dose indicate the iron is quite effective for a thickness up to 20 cm. As the thickness increases beyond 20 cm the apparent relaxation length becomes very long and eventually becomes longer than that for water, which causes the dose at a fixed distance from the source to increase with replacement of water with iron. The neutron dosage measurements were taken to aid in estimating the relative transmitted dose and radiation damage beyond the solid iron as compared with that in the boron carbide design.

62. Clifford, C. E.

GAMMA ATTENUATION OF LEAD IN TOP PLUG MOCK-UP NO. 1. Oak Ridge National Lab., Tenn. Rept. CF-52-5-1, Pt. 3, 26 June 52. Decl. with deletions 19 Apr 57, 4p. (Contract W-7405-eng-26)

An attempt was made to measure the gamma attenuation of lead in the beryllium-iron-water configuration. This configuration is a mockup of a top plug design which replaced all B_4C in the bottom section of the plug with solid iron. The iron was followed by sufficient water to remove most of the low-energy neutrons, which would cause capture gammas in the subsequent lead gamma shielding. Neutron measurements indicated the amount of water used was sufficient. Unfortunately the observed gamma intensity was too low to allow reliable measurements of the attenuation of more than one slab; however, the data for two slabs are presented as obtained.

63. Clifford, C. E.

NEUTRON AND GAMMA DOSE DISTRIBUTION BEYOND VARIOUS SECTIONS OF TOP PLUG MOCKUP NO. 2. (BERYLLIUM, SOLID IRON, BORON CARBIDE, WATER, AND IRON-WATER MIXTURE). Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-1, Pt. 4, 6 Aug 52. Decl. with deletions 17 Apr 57, 11p.

63. (cont'd) (Contract W-7405-eng-26)

Top plug mockup 2 was the first experimental test of the initial top plug design using B_4C in a thick layer in the high-temperature region of the plug. Thick mockup included the 20-in. B_4C slab which contained occluded water in an unknown amount due to improper storage. Since the water is very effective in removing the intermediate energy neutrons which would normally have penetrated the B_4C and would have been captured in the iron layers following it, the mockup gives an underestimate of the capture gamma source in the iron-water layers. An attempt was made to estimate the effect of the water in the 20-in. B_4C slab by eliminating water from between the iron slabs. Since the iron appears to be as effective as could be expected and no increase in the gamma dose was noted upon the elimination of 20 cm of water in the iron-water mixture, it is concluded that capture gammas in this design are not a significant fraction of the measured gamma dose. The measurement of the gamma dose was very difficult since the radiation from the source was less than the background. The neutron dose could not be measured directly with the dosimeter because of the low intensity. The thermal flux was used to estimate the fast neutron dose on the basis of a previous comparison with the dosimeter reading in a pure water shield where the neutron relaxation length was between 7 and 9 cm. Since other measurements have indicated that for neutron relaxation lengths as short as those prevalent beyond this mockup, this procedure overestimates the dose by roughly an order of magnitude; therefore, the reported dose is felt to be a conservative upper limit.

64. Clifford, C. E.

NEUTRON AND GAMMA DOSE DISTRIBUTION BEYOND PRIMARY SHIELD MOCK-UP NO. 2 (BERYLLIUM, SOLID IRON, WATER, AND SOLID IRON). Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-1, Pt. 5, 17 July 52. Decl. with deletions 19 Apr 57, 4p.

(Contract W-7405-eng-26)

Since the gamma attenuation of the first primary shield mockup was greater than required, a second mockup was tested which contained less iron. Also the B_4C layer was eliminated since it was a difficult material to handle in the shield and was not particularly advantageous at this location. The B_4C was initially present to reduce capture gamma production and radiation damage in the region requiring high-temperature shielding material, i.e., close to the reactor. It was found, however, that solid iron would suffice for this region and was preferred from the engineering point of view.

65. Clifford, C. E.

NEUTRON AND GAMMA DOSE DISTRIBUTION BEYOND TOP SHIELD FOR VARIOUS WATER THICKNESSES IN PRIMARY SHIELD MOCKUP NO. 2. Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-1, Pt. 7, 8 Aug 52. Decl. with deletions 19 May 57, 7p. (Contract W-7405-eng-26)

The neutron and gamma dosage distributions were measured as a function of the water thickness in the primary shield in order to estimate the contribution to the total dose by capture gammas produced within the top shield.

66. Clifford, C. E.

NEUTRON AND GAMMA DOSE DISTRIBUTION BEYOND VARIOUS SECTIONS OF TOP PLUG MOCKUP NO. 3. (BERYLLIUM, SOLID IRON, IRON-BORON CARBIDE MIXTURE, SOLID IRON). Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-1, Pt. 9, 17 July 52. Decl. with deletions 19 Apr 57, 11p. (Contract W-7405-eng-26)

Based on information obtained from earlier experiments and further engineering study, a final design of the top plug was made in which the B_4C layer was reduced in thickness and replaced with solid iron in the inner regions. The thickness of the iron and B_4C was chosen to reduce the radiation damage in the water following the layers to a tolerable limit. The actual shield design stipulated a 40% Fe-60% H_2O mixture for the region beyond the B_4C layers.

67. Clifford, C. E.

NEUTRON AND GAMMA DOSE DISTRIBUTION BEYOND TOP PLUG MOCKUP NO. 3 WITH MOCKUP OF STRUCTURAL MEMBER (SOLID IRON MODIFIED TO 47% IRON-53% WATER. Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-1, Pt. 10, 4 Aug 52. Decl. with deletions 17 Apr 57, 9p. (Contract W-7405-eng-26)

The design of the removable top shield plug necessarily incorporated rather massive structural members of solid steel which penetrated straight through the shield, thus allowing a leakage path for intermediate energy neutrons. A partial mockup of the worst case, the 4 in. shell of steel surrounding the plug, was installed in the Lid Tank to determine the order of magnitude of the leakage problem. The results of the test were not encouraging. The presence of the 4-in. solid steel layer increased the emerging flux by an order of magnitude over the unperforated shield for only 13 in. of perforation. Since the results of this experiment were not encouraging, a further investigation was made for many configurations of solid steel ducts in a water shield and are reported in Part II of this memorandum.

68. Clifford, C. E.

CONTROL PLUG MOCKUP MEASUREMENTS; NEUTRON AND GAMMA DISTRIBUTION IN WATER CONTAINING LEAD, STEEL, AND AIR. Oak Ridge National Lab., Tenn. Rept. no. CF-52-5-183, 26 May 52. Decl. with deletions 19 May 57, 7p.

Measurements were completed in the Lid Tank on the short experiment to aid in the design of control-instrument plugs. The plug was designed to increase the thermal-neutron flux locally in the water beyond the pressure shell. This was accomplished by penetrating the water reflector and the steel and water thermal shield with an 8-in. diameter steel pipe filled with lead, steel, or air. The measurements are compared

68. (cont'd) with those of the unperforated shield. Neutron measurements were taken in water along the axis of the plug and also along the line perpendicular to and intersecting the plug axis. Gamma measurements were taken beyond the air-filled plug only. Neutron-induced secondary rays are predominant in determining the attenuation rate of the pressure shell.

69. Cranford, W. and Miller, R. A.

Results of an optimization procedure.

Convair, Fort Worth, Tex. In Paper 10

of PAPERS FROM SEVENTH SEMIANNUAL

SHIELDING INFORMATION MEETING,

14-15 OCTOBER 1959. TID-6302, 10p.

Some results are reported on reactor and crew-compartment shields minimized by an optimization procedure called GYPSY. For a fast-neutron shield, the question of the adequacy of using a representative point source appears to be settled affirmatively. The effect of optimization of a crew shield for different point detector positions was investigated.

70. Curtis, H. J.

Limitations on space flight due to cosmic

radiations. SCIENCE v. 133, p. 312-16,

3 Feb 1961.

Study considering cosmic radiation in terms of a hazard to space missions. Origin and nature of cosmic radiation are reviewed, and radiation hazards are reduced to two zones around the earth (referred to as the Van Allen radiation belts), with maximum concentrations occurring in the plane of the magnetic equator, at peak frequencies of roughly 3,500 and 18,000 km. from the surface of the earth. A theory on the nature of these radiation belts is outlined, and estimates are made of peak intensity radiation doses for both radiation belts. Radiation that would be encountered inside a space ship passing through the belts is analyzed and the required shielding estimated. Finally the effects of primary cosmic rays, including the effect of thin-downs, on organs most likely to be affected are discussed. The study leads to the following conclusions: flight below the Van Allen belts seems reasonable safe without radiation shielding; a rocket shielding sufficient to permit a man to remain in the inner Van Allen belt for more than an hour is probably impractical, but passage through it should be possible without any serious harm to man; shielding for the outer Van Allen belt is possible but would have to be quite heavy if a stay of more than a few hours were contemplated; finally, the primary cosmic radiation is not intense enough to deliver a

70. (cont'd) serious radiation dose, even for exposures of a few weeks, and the heavy cosmic ray primaries do not seem to present an unusual hazard.

71. Dacey, J. E., et al

NUCLEAR SHIELDING STUDIES III

SHIELDING PROPERTIES OF VARIOUS

MATERIALS AGAINST NEUTRONS AND

GAMMA RAYS. Massachusetts Institute

of Technology, Laboratory for Nuclear

Science and Engineering, Cambridge, Mass.

Rept. no. 23; N5ori 07818. ASTIA ATI 194-021

Thermal and indium resonance (1.44 ev) neutron fluxes were determined with indium foils in aluminum and cadmium holders, and the gamma-ray ionization intensity was measured with pocket-type dosimeters, as functions of distance from a 1.03 gram Ra-Be source centrally located in a 4-foot cubical tank. The source was centered in 6-, 4-, and 2-inch radius spherical shells. The relaxation length, the mean square distance, the slowing-down distance, and the resonance escape probability were determined for various neutron distributions. Values for the thermal absorption cross sections of hydrogen and chlorine, and for the thermal diffusion lengths in water and brine, resulted from the diffusion data. Approximations were made of the effective linear absorption coefficients of the aqueous media in the tank, and of the effective linear and mass absorption coefficients of the heavy shielding materials in the sphere. On the basis of human tolerance dosages of both gamma-rays and neutrons, minimum composite shields (weight and size) were determined.

72. Danno, A., et al

OPTICAL PROPERTIES OF THE SHIELDING

WINDOWS OF Co-60 GAMMA RADIATION

FACILITY. Japan. Atomic Energy Research

Inst., Tokyo. Rept. no. JAERI-1005, 1959, 23p.

Shielding windows manufactured by Corning, Penberthy, and Nippon Kogaku Co. were examined. Bubbles, stones, and scratches were found in all three windows in considerable quantity, and striae were found in the windows of Penberthy and Nippon Kogaku. Transmittances of the windows measured for D-line were about 25% for Nippon Kogaku's, 25% for Corning's and 14% for Penberthy's.

73. Dennis, R., Purohit, S. N. and Brownell, L. E.

PROCEDURES FOR SHIELDING CALCULATIONS.

Michigan. Univ., Ann Arbor. Engineering Research Inst. Technical rept. no. 1; ERI-1943-8-86-T, Jan 57, 112p. (Contract AT (11-1)-162)

The shielding of nuclear radiation facilities is studied. Gamma-radiation shielding and the concept of the "build-up factor" are discussed in detail. As an illustration, the analysis of the heterogeneous gamma-radiation spectrum from the MTR fuel element from the point of view of shielding is given. The analytical expression for the build-up factor, as obtained by Taylor, was used in the above analysis. The problems involved in shielding a nuclear reactor are discussed. A sample calculation of the shielding of a nuclear reactor is given. Graphs for determining the standard integrals involved in calculating the radiation flux for standard geometries and an extensive bibliography are included.

74. Desov, A. Ye.

Radiation protection. STROITEL. PROM. no. 2, p. 28-31, 1956. (Translated from REFERAT. ZHUR. KHIM. no. 3, 1957, Abstract no. 10713)

The absorption coefficient for Co^{60} γ radiations was measured for five concrete compositions using heavy filters and for cast iron. Slabs 75 by 75 by 10 cm and 75 by 75 by 3 cm were irradiated with γ-rays, using a broad and a narrow band. It was established that when a narrow beam (diameter of the diaphragm, 10 to 12 mm) is used the intensity of the radiation decreases more rapidly than when a broad beam is used. The absorption coefficient (in cm^{-1}) for narrow and broad beams are as follows: cast iron, 0.42 and 0.289; concrete with various fillers: limonite and cast iron, 0.249 and 0.205; sand and cast iron, 0.22 and 0.186; limonite and limonite, 0.152 and 0.136; sand and gravel, 0.134 and 0.123; sand and gravel with the addition of borax, 0.135 and 0.122. Also included are a nomogram for finding the shielding thickness as a function of the desired degree of absorption of the rays, data on the composition of the concretes investigated, diagrams of the apparatus, and attenuation curves for Co^{60} radiation.

75. De Vries, T.W.

ANALYSIS OF RADIATION-INDUCED MELTING OF A LEAD SHIELD. Convair, Fort Worth, Texas. Rept. no. NARF-60-26T; MR-N-245, 8 Aug 60, 42p. (Contract AF33 (600)-38946)

A numerical method for obtaining the thermodynamic steady-state solution of a lead shield being heated from within by gamma-ray absorption and cooled by radiation and natural convection is described. The method can be used to predict radiation-heat damage to shield materials. A comparison of calculated with observed results is included.

76. Division of Reactor Development, AEC.

REACTOR SHIELDING INFORMATION MEETING, CHICAGO, 12-13 NOVEMBER 1953. Rept. WASH-152, Mar 54. Decl. with deletions 6 Mar 57, 106p.

Thirty-one papers are presented on reactor shielding design and development including experimental and theoretical problems. The titles are: Neutron Capture Gamma-Ray Spectra; Gamma-Ray Streaming through Cylindrical Voids in Water; The Effects of Impurities on the Streaming of Neutrons through Iron; The Transmission of Na^{24} Gamma-Rays through Lucite and Steel for Several Angles of Incidence; The NDA Shielding Programs; Neutron Penetration in Hydrogen and the Removal Cross Section Concept; Experimental Breeder Reactor Shield Measurements; Monte Carlo Studies; Solution of Gamma-Ray Diffusion Problems by Random Sampling Methods; Discussion of Certain Boundary Effects Upon the Propagation of Gamma Rays; Gamma Radiation from Inelastic Scattering of 3.1-Mev Neutrons; Analysis of the Scintillation Spectrometer Observations of the Penetration of Cs^{137} Gamma Radiation through Water; Fast Neutron Removal Cross Sections; Neutron Spectra in Water; Slant Penetration of Neutrons in Water; Attenuation of Neutrons through Thick Shields; Ground Scattering of Co^{60} Gamma Rays; Analysis of Experimental Scattering of D (d,n) Neutrons from a Plane of Aluminum; Gamma Spectrum of the Bulk Shielding Reactor; Prompt Fission Gamma Rays; Broad Beam Gamma Attenuations; Lithium Iodide Crystals; Air Duct Tests; the Semi-Analytical Method of Shield Design; Reactor Leakage for Shielding Calculations; Fast neutron Dosimetry; Capture Gamma Ray Studies; Energy Effects in Reactor Shields; Numerical Studies of Non-Axial Distributions from Disk Sources of Radiation; Calculation of the Bremsstrahlung from Li^8 Electrons.

77. Donnert, H. J.

NEUTRON TRANSPORT STUDIES.

Army Chemical Corps. Nuclear Defense Lab., Army Chemical Center, Md.

Rept. no. NP-10038.

An outline is given of neutron transport studies conducted toward the solution of the source data problem for radiation environments caused by nuclear weapon detonations in air and at high altitudes.

78. Donovan, L. K. and Chilton, A. B.

DOSE ATTENUATION FACTORS FOR CONCRETE SLAB SHIELDS COVERED WITH FALLOUT AS A FUNCTION OF TIME AFTER FISSION. Naval Civil Engineering Lab., Port Hueneme, Calif. Technical rept. no. 137; NP-10306, 1 June 61, 31p.

A study was made to determine the dose attenuation of fallout gamma radiation by various thicknesses of concrete roofs of buried personnel shelters. Dose attenuation factors are derived as a function of time after a nuclear detonation.

79. Dubois, F.

DESIGN AND INSTALLATION OF CONCRETE SHIELDING IN THE CIVIL ENGINEERING OF NUCLEAR CONSTRUCTIONS. (Étude et mise en place des bétons de protection dans le genie civil des ouvrages nucleaires). France. Commissariat à l'Energie Atomique. Centre d'Etudes Nucléaires, Saclay) 1960, 165p.

79. (cont'd) Technical information about high density concretes which are very important for radiation biological shielding is presented. The heavy aggregates (barytes, ilmenite, ferrophosphore, limonite, magnetite and iron punchings) used to make these concretes were investigated from the point of view of prospecting and physical and chemical characteristics. A general survey of shielding concretes was made involving the study of components and mixing and placing methods. A detailed investigation was also made of some high-density concretes: barytes concrete, with incorporation of iron punchings or iron shot, ferrophosphore concrete, ilmenite concrete, and magnetite concrete, more particularly with regard to grading and mix proportions and testing process. Two practical designs are described. Specifications are given for barytes concretes and for making the proton-synchrotron "Saturne" shielding blocks.

80. Dubois, F.

Heavy concretes with baryte and iron ore base. BULL. INFORM. SCI. ET TECH. (PARIS) n. 36, p. 2-24, Jan 1960. (In French)

The development of nuclear energy has led to the increasing importance of shielding against radiation. Concrete has been considered one of the best materials. However, the density of ordinary concrete is often insufficient. The use of high-density concrete as radiation shielding is discussed. The characteristics of shielding concretes are tabulated with a description of baryte concretes with iron scraps incorporated and of iron ore concretes. The biological shielding of Saturn and of very hot cells is discussed as two examples of the utilization of heavy concretes.

81. Duchêne, J.

IMPROVEMENTS IN SHIELDS FOR PROTECTION AGAINST NEUTRONS. (To Commissariat à l'Énergie Atomique). BRITISH PATENT 816, 235. 8 July 59.

A flexible shielding material and its preparation are described. Boron carbide powder is mixed with a polyvinyl chloride resin dissolved in a suitable solvent. The solvent is evaporated and the mixture cast in a mold shaped like a glove, suit, cr other shape for protection of a worker. Heating at 150° C for 1 hr. solidifies the material.

82. Dukleth, E. A., et al

DESIGN AND CONSTRUCTION OF THE HIGH DENSITY CONCRETE SHIELDING FOR THE ENGINEERING TEST REACTOR. Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. Rept. no. AECU-3655, 15 May 57, 57p. (Contract AT(10-1)-770)

Economic studies showed that high-density concrete made with magnetite was the most feasible choice from a cost and construction standpoint. Except for a few precautions, this concrete can be placed in essentially the same manner as conventional concrete. Concrete of any strength can be made, within reason, but for concretes having strengths of less than 4,000 psi at 28 days, extra fines must be added to the mix to lend supporting power and lubricity to the paste. Modulus of elasticity and Poisson's ratio for this dense concrete are approximately 50% higher than for normal-weight concretes of the same compressive strength.

83. Duncan, D. S. and Speir, A. B.

GRACE II- AN IBM 709 PROGRAM FOR COMPUTING GAMMA-RAY ATTENUATION AND HEATING IN CYLINDRICAL AND SPHERICAL GEOMETRIES. North American Aviation, Inc., Atomics International Div., Canoga Park, Calif. Rept. no. NAA-SR-Memo-4649, 12 Nov 59, 61p.

GRACE II is a multigroup, multiregion, gamma-ray attenuation code written in FORTRAN II for the IBM 709. The code is used to compute the total dose rate or heat generation rate from either a spherical or a cylindrical source. The source may be located in either the central region of the system or in a concentric shell region surrounding it. The source distribution may be uniform, exponential or have a polynomial variation in the radial direction, and, in the case of cylindrical geometry, may also have a polynomial variation in the axial direction. A least squares fit is used

83. (cont'd) to btain the polynomial coefficients. The buildup factor is represented analytically in a double exponential form. The shield regions outside the source volume may be either semi-infinite slabs or concretic shells. As many as 22 regions, 10 mesh points per region, 20 gamma-ray energy groups, 20 shield materials, and 20 material buildup factors may be included in a single calculation.

84. Dunn, C. D., et al

ACTIVATION HANDBOOK FOR AIRCRAFT DESIGNERS VOL. II. Convair, Fort Worth, Texas. Rept. no. NARF 55-55T; FZK-9-089-2, 31 July 57.

The present volume of the Activation Handbook for Aircraft Designers, Data Table for Activation Calculations, is designed to provide the information necessary for the calculation of activation gamma dose rates from the individual radioisotopes of any material. Data are included for 220 isotopic neutron reactions leading to the formation of radioisotopes. A comprehensive survey of the literature has been made to incorporate the best data available at this time.

85. Ebert, H. G.

Measurements of dose build-up factors for radiation shielding. (Institut fur Medizinische Physik und Biophysik, Göttingen, Germany). Z. AGNEW PHYS. v. 13, p. 95-9, Feb 1961. (In German)

For the calculation of a shielding wall for radiation protection not only the attenuation factor μ but also the dose build-up factor are necessary because of the expansion of the wall and the resulting increase of the dose output. Three build-up factors with various capacity can be distinguished: (a) The dose build-up factor was calculated for a wall of arbitrary size. No universal validity can be claimed for this build-up factor; it is related only to a special measurement apparatus. (b) The dose build-up factor was calculated for an infinite expanding medium in which radiator and absorber are found (B^+). Values of this kind are published in the literature, but with an accuracy of only = 10 to 20%. (c) The dose build-up factor was calculated for a wall of infinite size. But this value B_∞ is significant for radiation shielding since through it the maximum dose output obtainable behind a shield can be calculated. A measurement method was investigated and described which permits the measurement of B_∞ with great accuracy (±5%) on relatively small absorber pieces. This measurement method

85. (cont'd) is suitable for radiation shielding materials of high and average atomic numbers. The measurement results agree, in the range investigated, with the B^+ values given in the literature. For materials with low atomic numbers, deviations occur. Values for B_{∞} are given for various materials and radiation qualities.

86. Eckels, T. W.

MEASUREMENT OF LIGHT TRANSMITTANCE THROUGH THICK SHIELDING WINDOWS. Argonne National Lab., Ill. Rept. no. ANL-6159, May 60, 16p. (Contract W-31-109 eng-38)

A method was developed for measuring the light transmittance of thick shielding windows. The measuring equipment was mounted on the operator side of the window and light passed from outside the cell through a cylindrical access hole in the rear wall opposite the window being measured. These measurements were unaffected by the level of room lighting or movements of personnel in light-colored clothing. The results were reproducible within 2%. Supporting experiments verified the general accuracy of the method and equipment.

87. Edmonson, N., Jr., Henrick, J. J., and Moss, T. A.

Application of the Carlson S_n-method to shield calculations. Convair, Fort Worth, Tex. In Paper 9 of PAPERS FROM SEVENTH SEMI-ANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER, 1959. TID-6302, 15p.

The Carlson S_n-method is being adapted to the calculation of angular and total flux distributions in shielding configurations. Accordingly, computational procedures for spherically symmetric shields were developed which accounts for the anisotropy of elastic scattering of neutrons. Preliminary calculations for a water shield gave flux distributions that compare favorably with the results of moments methods and Monte Carlo calculations.

88. Edmonson, N., Henrick, J. J. and Moss, T. A.

APPLICATION OF THE CARLSON S_n-METHOD TO CALCULATION OF NEUTRON ANGULAR AND TOTAL FLUX DISTRIBUTIONS IN A SPHERICALLY SYMMETRIC SHIELD.

Convair, Fort Worth, Texas. Rept. no. NARF-60-11T; MR-N-253, 15 May 60, 121p.

(Contract AF33(600)-38946)

The S_n-method for calculating numerical approximations to solutions of the Boltzmann transport equation is developed for a spherically symmetric shield system. Anisotropic elastic scattering is assumed. Energy-group transfer coefficients are developed by using the Dirac σ-function and by a mapping procedure. Formulas for transforming cross-section data from the center-of-mass coordinate system to the laboratory coordinate system are given. A general matrix formulation for the anisotropic case is constructed. The procedures are applied to the computation of angular neutron distributions and total neutron distributions in water shields 80 and 120 cm in radius for Watt fission sources uniformly distributed in a small sphere concentric with the water shields.

89. Edwards, W. E. and Loechler, J. J.

SHIELDING COMPUTER PROGRAM 12-0 GAMMA RAY PENETRATION POINT SOURCE. General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinatti.

Rept. no. APEX-404, Aug 58, 22p.

(Contracts AF 33(038)-21102 and AT(11-1) -171)

Shielding computer program 12-0, which computes the gamma-ray flux, energy absorption rate, or dose rate from an isotropic point source, is coded for the IBM 704 computer. One frame of magnetic core memory and two tape units for utility routines are required. No magnetic-drum memory is required.

90. Eisenhauer, C.

METHOD FOR EVALUATING PROTECTION AFFORDED BY STRUCTURES AGAINST FALLOUT RADIATION. National Bureau of Standards, Washington, D. C. Rept. no. NP-100038, p. 138-52.

Considerations are given for the dose rate three feet above an infinite smooth plane contaminated with radioactive fall-out, and the corresponding angular distribution inside a structure which has heavy walls and contains some windows. Diagrams are included for angular distributions in open fields, in structures with windows, for thin and thick walls, and for a detector at window-sill level. Reduction factors for radiation from ground sources around, and roof sources on the blockhouse.

91. Emslie, A. G.

GAS CONDUCTION PROBLEM WITH MULTI-LAYERED RADIATION SHIELDS. Arthur D. Little, Inc., Cambridge, Mass. Rept. no. 63270-04-01, Apr 61.

On a long mission in space, a cryogenic fuel tank may require radiation shielding consisting of 100 sheets of low-emissivity metal foil if heat is transferred through the shielding only by radiation. If gas conduction also occurs, more foils will be needed for the same rate of fuel boil-off. For a gas pressure of 1.4×10^{-4}mm Hg, 200 foils are required. Consequently, an adequate sealed-off, evacuated shield is difficult to construct. If outgassing of the foils and gas diffusion from the tank are appreciable, it is also difficult to arrange the geometry of the shields to use the external space vacuum for pumping. In the case of pumping through the edges of a 100-cm-wide shield panel, the outgassing rate should not exceed about 10^{0} mol. sec^{-1} from each cm^2 of foil surface, if the number of foils is to remain about 100. The allowable outgassing rate for broadside pumping of optimally perforated foils is around 10^{10} mol. sec^{-1} cm^{-2}. The allowable diffusion rate from the fuel tank is 10^{12} mol. sec^{-1} cm^{-2} of tank surface. A basic consideration is that any geometrical arrangement of the foils that enhances pumping of the gas also reduces the effectiveness of the foils as a radiation shield, since radiation can enter by the same path by which molecules leave. As a result, only a limited number of layers of foil is useful. Beyond this number, no further improvement in shielding is gained.

92. Emslie, A. G.

RADIATION TRANSFER BY CLOSELY SPACED SHIELDS. Arthur D. Little Inc., Cambridge, Mass. Rept. no. 63270-04-02, May 61.

The usual formula for radiation transfer through a stack of radiation shields breaks down when the spacing of the shields is less than the wavelength of the peak of the black-body spectral distribution corresponding to the temperature of the shields. Two effects set in at these close spacings - wave interference and radiation tunneling. Wave interference of the emitted radiation occurs in the narrow gaps between the shields and may increase or decrease the energy transfer, depending on the spacing. Radiation tunneling allows transfer of radiation that ordinarily suffers total internal reflection inside the shield material. This effect gives an energy transfer that increases exponentially as the spacing decreases. The two effects together give an energy transfer rate per unit area which becomes, in the limit of zero spacing,

$$q = \frac{n^4}{n^2 + k^2} \sigma (T_2^4 - T_1^4)$$

where n and k are the real and imaginary parts of the complex refractive index, σ is the Stefan-Boltzmann constant, and T_2 and T_1 are the temperatures on the two sides of a gap. The formula implies that the radiation density e' and velocity of propagation σ' in the shield material are:

$$\sigma' = \frac{n^2 \sigma T^4}{c} \qquad \sigma' = \frac{n^2 c}{n^2 + k^2}$$

For moderate values of the absorption index k, the flux formula predicts a transfer rate between two close shields greater than that between two black surfaces. In the case of metal shields, when the spacing between the two shields is increased from zero, the radiation transfer rate at first rises sharply to a high maximum and then falls below the usual value for widely spaced shields. The flux returns to the normal level when the spacing exceeds about one half of the wavelength of the black-body peak.

93. Epstein, H. M., Dingee, D. A. and Chastain, J. W.

A STUDY OF THE PRIMARY SHIELD FOR THE PRDC REACTOR. Battle Memorial Inst., Columbus, Ohio. (For Atomic Power Development Associates, Inc.) Rept. no. BMI-APDA-622, 15 Apr 57, 34p.

93. (cont'd) Temperature distributions, irradiation effects, stacking arrangements, voidage, and economics for the borated-graphite shield of the PRDC reactor were investigated. Of the shield systems considered, four are reported here. System 1 contains 30 in. of 1% borated graphite, with either ordinary graphite or a cement as a filler for the remainder of the volume. The maximum temperature at the flex plates in this system was calculated to be 500°F. Systems 2 and 3 consist of 2 in. of 5% borated graphite near the core vessel and 1/2 in. of Boral at the primary-shield tank. A filler material of carbon blocks is used in System 2 and graphite in System 3. The calculated maximum temperatures were 700°F and 350°F, respectively. System 4 consists of a laminated structure of Boral and graphite near the primary-shield tank and carbon-block filler. It was calculated to have a maximum temperature of 600°F at the flex plates. The maximum temperature at the flex plates recommended by APDA is 500°F. Energy storage and radiation damage were found to be within permissible limits in all four systems. However, these conclusions are based on experimental data from the Hanford reactor in which the neutron-energy spectrum differs considerably from the PRDC spectrum. A porosity of less than 740 cu ft. is required in order that a sodium leak from the core vessel does not expose the core. The voidages in any of the systems mentioned above is about 400 cu ft. excluding absorption effects. These are believed to be small. The systems containing Boral were found to be less expensive than the ones using only borated graphite. Over-all material costs range between $230,000 for Boral systems and $350,000 for borated-graphite systems.

94. Espe, W.

On neutron penetrance of industrial glasses.

STROJNOELEKTROTECHNICKY CASOPIS

v. 9, n. 3, p. 168-71, 1958. (Translated from

REFERAT. ZHUR. KHIM. n. 2, Abs. 5494, 1959)

Analyzing the coefficient of specific absorption of neutrons (Ns) by various oxides taking part in the composition of glass (G), it was concluded that ordinary industrial Gs are characterized by a great neutron penetrance. On the contrary, G containing oxides of Cd, B, Gd and Eu, absorbs neutrons very efficiently. Indium oxide possesses an exclusively great capacity of selective absorption of neutrons of an energy of 1.4 electron-volt. Recipes for, and properties of, some Gs designed for protection from neutrons are presented.

95. Evans, T. C.

RADIATION HAZARDS OF SPACE EXPLORATIONS AND RADIOBIOLOGICAL PRINCIPLES INVOLVED. Space-nuclear Conference,

95. (cont'd) Gatlinburg, Tennessee, 3-5 May 1961, p. 1756-61.

The types and quantities of radiations to be encountered in space exploration are numerous and varied. Radiobiological principles involved are (1) total absorbed dose, (2) dose rate, (3) volume dose, and (4) relative effectiveness of different kinds of radiation. The degree and type of biological damage will vary, with changes in the above-mentioned factors, from immediate and definite to late and indeterminant. It would appear at this stage of the study that biological hazards of radiations will not be a limiting factor in comparison with the many other risks involved.

96. Fasano, A. N. SOME NEW TYPES OF NEUTRON-SHIELDING MATERIALS. Wright Air Development Center. Air Force Nuclear Engineering Test Facility, Wright-Patterson AFB, Ohio. Rept. no. WADC-TN-59-227; Proj. no. 7112, June 59, 6p.

Research reported was conducted between 1954 and 1957 at Brookhaven National Lab. Materials having high neutron-absorption cross sections are needed for protecting gamma and neutron detectors from thermal-neutron damage and to form collimators for beams of neutrons. Boron or lithium, as the powdery compounds B_4C and LiF could absorb the neutrons, but as powders are impossible to apply or form into protective shields. Processes were developed using inexpensive plastic, paraffin, and linseed oil as bonds and G. E. thinner, Saurizen thinner, Trichloroethylene, and turpentine as solvents to produce materials that can be molded, formed, cast, rolled, or painted as a protective shielding or coating. Methods were devised for reclaiming the expensive boron and lithium compounds.

97. Fenstermacher, C. and Henshall, J. RADIATION SHIELDING FOR TEST CELL "C". Los Alamos Scientific Lab., N. Mex. Rept. no. LAMS-2474, June 59, 25p. (Contract W-7405-eng-36)

97. (cont'd) Information is presented, related to radiation shielding at test cell "C," for calculations of wall and roof thicknesses, radiation heating of front wall by gamma and neutron absorption, neutron activation in front face of test cell, γ-dose rates both direct and behind shadow shields at various distances, and dose produced by a reactor catastrophe.

98. FitzSimons, N.

ENGINEERING APPLICATIONS OF FALLOUT GAMMA RADIATION SHIELDING RESEARCH. Office of Civil and Defense Mobilization, Battle Creek, Mich. Rept. no. NP-10038, p. 153-72.

Functional equations with related graphs were developed for solving common shelter shielding problems. Examples were worked out to illustrate procedures for analyses involving both above- and below-ground shelters, as well as aperture problems.

99. Flew, E. M. and James, B. T.

CALCULATIONS OF U, Pb, Fe AND Al SHIELDING FOR IRRADIATED NATURAL URANIUM. United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England. Rept. no. AERE-HP/GEN-6, 5 Dec 55. Decl. 31 Mar 58, 6p.

A list of all gamma-emitting fission products with significant yield after decay for one day was compiled, and the products grouped according to gamma-energy ranges. Dosage transmission factors were calculated and graphs plotted for fractions of gamma dosage transmitted against thickness of aluminum, iron, lead, and uranium for cooling times of 1, 16, and 63 days assuming an irradiation time of 3 weeks.

100. Foelsche, T.

Estimate of the specific ionization caused by heavy cosmic ray primaries in tissue or water. JOURNAL OF THE ASTRONAUTICAL SCIENCES v. 6, n. 4, p. 57-62, 1959.

The purpose of the calculations is to estimate the specific ionization caused by heavy cosmic ray primaries along their paths in living tissue or in water.

101. Fraas, A. P.

Nuclear aircraft shielding. AERONAUTICAL ENGINEERING REVIEW v. 15, n. 9, p. 39-43, Sep 1956.

Types of radiation from nuclear reactors and their effects are discussed in relation to shielding materials and configurations for aircraft applications.

102. French, R. L.

REACTOR SHIELD HEATING CALCULATIONS FOR GAMMA RAYS. Convair, Fort Worth, Tex. Rept. no. FZM-1076, June 58, 28p.

The theory of γ heat generation is discussed, and a new method for calculating heat-generation rates is presented. The method employs differential-energy spectra in an iteration formula which gives the modified γ spectrum from an elemental volume of the reactor core after penetration of successive layers of material. An integration over the core volume gives the total γ spectrum for the point at which the heat-generation rate is to be determined. Application of energy-absorption coefficients and an integration of the spectrum over energy yield the heat-generation rate. Results obtained with the method agree within 30% with heat-generation rates measured at the Bulk Shielding Reactor.

103. Frynta, Z. and Langmajer, J.

Containers for Co^{60}. Inst. of Materials and Tech., Prague. JADERNA ENERGIE v. 4, p. 98-102, Apr 1958. (In Czech)

Data are given for the determination of the wall thickness in the design of lead containers encased in thick iron encasements. The performance and practical factors of such lead-iron containers for high activity Co^{60} are discussed.

104. Gabro, A. N. and Mooney, L. G.

ASTR FAST-NEUTRON AND GAMMA-RAY SPECTRAL MEASUREMENTS. Convair, Fort Worth, Texas. Rept. no. NARF-60-21T; MR-N-263, 15 Jan 61, 64p. (Contract AF33(600)-38946)

Gamma and fast-neutron radiation spectra from the Airborne Shield Test Reactor (ASTR) are determined. Gamma-ray spectral measurements are presented for the following experimental arrangements: (1) In a horizontal plane through the centerline of the ASTR, at $\theta = 0^o$ through 180^o, at a 63-ft. separation distance, for ASTR configurations 3 with the boral cover on the ASTR and the ASTR shield water borated, and using a collimated detector; and (2) in the small cylinder (geometry D), at $\theta = 0^o$ at a 33-ft. separation distance, for ASTR configuration 3 with the boral cover off the ASTR and the ASTR shield water not borated, and using a bare-crystal uncollimated detector. Fast neutron spectral measurements were made in a horizontal plane through the centerline of the ASTR at a 33-ft. separation distance, and are presented for: (1) $\theta = 0$, 105, and 120^o for ASTR configuration 3; (2) $\theta = 105^o$ for ASTR configuration 5; and (3) $\theta = 90^o$ for ASTR configuration 15.

105. Gage, A. M.

NEUTRON FLUXES, GAMMA-RAY DOSE RATES, AND TEMPERATURES IN THE IRON-MAGNETITE CONCRETE SHIELD OF THE OMEGA WEST REACTOR. Los Alamos Scientific Lab., N. Mexico. Rept. no. LA-2155,

105. (cont'd) 7 Oct 57, 84p. (Contract W-7405-eng-36)

Thermal and fast neutron fluxes were measured in the shield of the Omega West Reactor at Los Alamos with In foils and S pellets, respectively. The In foils and β counters were calibrated to a known flux of thermal neutrons in a graphite sigma pile. The S pellets and counters were calibrated to a known flux of 14.1 Mev neutrons from a T(d,n) He source. The measured effective decade length for the fast neutron flux is 8.1 in. The average Cd ratio is 2.4, varying from 2.1 to 2.6. The distribution of γ-ray dose rates, measured with dosimeters, has a decade length of 7.0 in. At the outer surface of the shield (4 ft of water and 5 ft of concrete) the fast neutron leakage is 0.1 neutrons/cm^2-sec and the γ -ray dose rate is 2.0 mr/hr for a reactor power of 10 Mw (extrapolated from measurements made at 840 kw). No heating of the concrete by γ rays and neutrons was observed at a power of 840 kw. Small anomalies in the distribution of temperature, thermal neutron flux, and Cd ratio are consistent with one another and suggest variations in concrete density. The Fe-magnetite concrete has an average specific gravity of 4.6 and contains (by weight): 75% Fe, 19% 0, 3% Ca, 2% Si, 0.4% H, and 0.3% Al.

106. Ganguly, N.K.

Shielding manned space vehicles from space radiations. JOURNAL OF THE BRITISH INTERPLANETARY SOCIETY v. 18, n. 3, p. 110-14, May-June 1961.

The Van Allen belts, the steady-state values of primary cosmic radiation, and the time-variational part of the cosmic rays are discussed as contributing factors to the problem of designing adequate shielding for manned space vehicles.

107. General Electric Co., Aircraft Nuclear Propulsion Dept., Cincinatti.

PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. Rept. no. TID-6302; XDC-60-6-70, 226p.

Twenty-one papers were presented on radiation shielding. Separate abstracts were prepared for each paper.

108. Gerber, R. C., Jr.

GENERAL SHIELD THICKNESS EQUATIONS AND CURVES. Knolls Atomic Power Lab., Schenectady, N. Y. Rept. KAPL-M-RCG-2, 1 Feb 50. Decl. 28 Feb 57, 17p. (Contract W-31-109-eng-52)

The method of calculating shield thickness currently in use at KAPL is described, and some equations were plotted to facilitate future work.

109. Girolimetto, M.

Material absorbing ionising radiations.

BRITISH PATENT 871,521, 28 June 1961.

A material for the formation of screens substantially opaque to ionizing radiations such as x rays is described. The material consists of a thermoplastic or thermosetting polymer resin incorporating at least 5% by weight of mercuric iodide and at least 5% of the iodide of an alkali metal of Group I of the Periodic Table. The two iodides produce a double salt $3HgI_2 \cdot 2XI$ (where X represents the alkali metal). The resin in polymethyl methacrylate and is transparent to light in the molded form. The composition of the material may include the usual plasticizers and accelerators.

110. Gandusio, G. and Polezza, S.

Absorption cross section of carbon and polycrystalline graphite. (Section efficace d'absorption du carbone et du graphite poly-cristallin). CEA-tr-I-34. Translated into French by L. Roulet from ENERGIA NUCLEARIA (MILAN) v. 5, p. 383-6, 15p., 1958.

A review is presented of the methods and results of measurement of the absorption cross sections of polycrystalline graphite in a thermal neutron flux. It is concluded that noticeable margins of error still exist in such measurements. The pile-

110. (cont'd) oscillator method offers the best results, and a value of 3.2 mb for ideal carbon has the greatest probability of being exact.

111. Goetz, C. A. and Boling, M. A.
SCARF 1 (SCATTERING CONTRIBUTION FROM AN ARRAY OF RADIATOR FINS): A FIRST ORDER APPROXIMATION OF THE SCATTERED FAST NEUTRON CURRENT FOR SNAP REACTOR SYSTEMS. Atomics International. Div. of North American Aviation, Inc., Canoga Park, Calif. Rept. no. NAA-SR-Memo-5948, 8 Dec 60, 38p.

A description is given of SCARF-1 which is a computational aid designed for application to shielding problems involving spacecraft powered by SNAP systems. Specific applications of the code are outlined.

112. Gold, T.
Origin of the radiation near the earth discovered by means of satellites. NATURE v. 183, n. 4658, p. 355-58, 7 Feb 1959.

The energy range of the charged particles concerned seems to extend from perhaps several tens of kev to the several tens of Mev. Intensities of the order of a few hundred particles per steradian per sec. capable of penetrating the shielding of 2.5 gm per cm^2 have been recorded.

113. Goldstein, H.

THE ATTENUATION OF GAMMA RAYS AND NEUTRONS IN REACTOR SHIELDS. U.S. Atomic Energy Commission, Washington, D.C. 1957, 306p.

Available from U.S. Government Printing Office. A discussion of the fundamentals of shielding is presented; the factors affecting the permissible radiation levels, the sources and characteristics of the radiation to be shielded against, and bulk shielding measurements are discussed. The theoretical or empirical calculation of the attenuation of neutrons and γ rays in shield materials is discussed at length.

114. Goldstein, H. and Aronson, R.

ATTENUATION OF LiH REACTOR SHIELD PLUG. Nuclear Development Associates, Inc., White Plains, N.Y. Rept. no. NDA-Memo-14-28, 9 Mar 54. Decl. 28 May 58, 9p.

Work performed under contract with the Detroit Edison Co. and the Dow Chemical Co. A preliminary calculation was made of the attenuation of neutrons through a lithium hydride shield plug. It has been proposed that a neutron shield be placed between the reactor core and blanket to prevent the fuel element handling mechanism from becoming so active that it cannot be repaired in case of damage. These approximate results indicate that a plug with a thickness of approximately 70 cm will reduce the neutron current by 10^{-6}.

115. Gorshkov, G. V. and Kodyukov, V. M.

Attenuation of the γ-rays from sources extended in three dimensions by lead and iron. ATOMNAYA ENERG. v. 9, p. 139, Aug 1960. (In Russian)

Experiments with metallic containers filled with aqueous solutions of colloidal gold (Au^{198}, $E_2 = 0.411$ Mev) and NaCl (Na^{24}, E = 1.38 Mev and $E_2 = 2.76$ Mev) showed

115. (cont'd) that at given geometries the attenuation of γ rays from extended sources is similar tc the attenuation for a point source when the build-up factor is $(V = K_{theor.}/K_{exp.})$. The build-up factor for the extended sources is less than for the point source and depends on the shape. The buildup factor increases with reduced γ energy and decreases with the increase of the absorber atomic number. The increase in a three-dimensional-extended source in a lead absorber is close to 1.0 at μ 1 $<$ 3.

116. Grantham, W. J.

A STUDY OF CONCRETE SAMPLES

TAKEN FROM THE TOP SHIELD OF

THE ORNL GRAPHITE REACTOR.

Oak Ridge National Lab., Tennessee.

Rept. no. ORNL-3016, p. 245-6.

Samples secured duri ng the drilling of experiment holes 70 and 71 in the ORNL Graphite Reactor were analyzed for water content and compressive strength. At the time of the drilling, the reactor had developed a total power of 4.3 x 18^8 kwhr. Results indicated a water content consistent with design anticipations, but the compressive strength of both the barytes-haydite and ordinary concretes fell significantly below the design strength of 2.5 ksi.

117. Graves, G. L.

SOME FOIL ABSORPTION CALCULATIONS.

University of California, Los Alamos Scientific

Lab. Rept. no. LA 1964, June 55.

Calculations dealing with the transmission and absorption of neutron fluxes in single thin foils and in individual members of a foil sandwich consisting of three foils in the same material placed in intimate contacts are given.

118. Great Britain, Building Research Station,

Garston, Watford, Herts, England.

CONCRETE OF HIGH DENSITY FOR SHIELDING

ATOMIC REACTORS. Rept. no. AERE-RE/R-1406;

118. (cont'd) BRS-DR-40, Feb 54. Decl. 2 July 57, 20p.

A program of tests and development work on concrete for use in shielding atomic reactors was undertaken. Concretes made with river gravel barytes and steel shot aggregate were tested. Determinations were made of density, hydrogen content, strength, and thermal conductivity. Data are tabulated.

119. Greene, J. C.

OCDM INTEREST IN NUCLEAR RADIATION SHIELDING. Office of Civil and Defense Mobilization, Battle Creek, Mich. Rept. no. NP-10038, p. 100-26.

A discussion is given of the applications of shielding information in OCDM programs which include, inventorying existing fallout protection, designing protective shelters and war gaming. An analytical summary is given of structures and shelter spaces determined in a fall-out shelter survey for the Tulsa (Oklahoma) central business district. Designs of several protective shelters are illustrated for family and schools. The applications to war gaming are discussed according to damage assessments and risk computations.

120. Grotenhuis, M. and Butler, J. W.

CP-10 SHIELD DESIGN. Argonne National Lab., Lemont, Ill. Rept. no. ANL-4864, 1 June 52. Decl. 12 Feb 57, 37p. (Contract W-31-109-eng-38)

The design calculations for a reactor shield consisting of 2 ft of graphite, a sheet of boral, 6 in. of Fe, and 7.25 ft of ordinary concrete are given.

121. Gusev, N. G. and Kovalev, E. E.

NOMOGRAMS FOR COMPUTING SHIELDING REQUIREMENTS AGAINST Ra, Co^{60}, Cs^{137}, AND Ir^{192} γ RAYS. (<u>Nomogrammy dlya</u> rascheta zashchity ot gamma-lucheĭ ra,

121. (cont'd) $\underline{Co^{60}, Cs^{137}, i\ Ir^{192}}$. Moscow, Publishing House of the Main Administration on Atomic Energy, Council of Ministers, 1959, 71p.

Forty-five nomograms are plotted for fast and accurate estimates of shielding against equilibrium and non-equilibrium decay products of radium, Co^{60}, Cs^{137}, and Ir^{192}. The shielding materials are lead, iron, lead glass, concrete, and water. The nomograms are plotted for wide γ beams. Descriptions and instructions for use are given. The book is designed for engineers, medics and other personnel handling γemitting isotopes.

122. Haffner, J. W., Loechler, J. J. and MacDonald, J. E. AN IBM 704 PROGRAM REPORT, AIRCRAFT NUCLEAR PROPULSION SHIELDING PROGRAM 10-0. General Electric Co., Aircraft Nuclear Propulsion Dept., Cincinnati. Rept. no. APEX-503, Mar 58, 36p. (Contracts AF33(600)-38062 and AT(11-1)-171)

Shielding Program 10-0 calculates the fast neutron dose rate at a shielded point detector due to both direct-beam and single-scattered radiation in a homogeneous, infinite medium from an anisotropic point source. The detector shield and the angular distribution associated with the point source are both assumed to be symmetric about the source-detector axis. The fast-neutron-source energy spectrum may be approximated by ten discrete energies. Exponential attenuation may be considered on either leg, as desired. The dose rate arising from the source spectrum is obtained by a summation of the dose rates computed for each initial energy and for the rear, side, and front of the detector shield.

123. Haffner, J. W., Loechler, J. J. and MacDonald, J. E. IBM 704 PROGRAM REPORT, AIRCRAFT NUCLEAR PROPULSION SHIELDING PROGRAM 09-0. General Electric Co. Aircraft Nuclear Propulsion Department, Cincinatti. Rept. no. APEX-533, Dec 59,

123. (cont'd) 32p. (Contracts AF33(600)-38062 and AT(11-1)-171)

Shielding program 09-0 calculates the fast neutron dose rate due to single-scattered radiation in a homogeneous infinite medium from an anisotropic point source at any specified, unshielded point detector. The fast-neutron-source energy spectrum may be approximated by 10 energy levels. Exponential attenuation may be considered on either leg as desired. The dose rate from the source spectrum is obtained by summation of the dose rates computed for each initial energy.

124. Haggmark, L. G.

SHIP SHIELDING FACTORS-COMPUTATIONAL METHOD COMPARED TO EXPERIMENTAL RESULTS. Naval Radiological Defense Lab., San Francisco, Calif. Rept. no. USNRDL-TR-514, 7 June 61, 24p.

A previously developed computational method was used to obtain the ratios of the gamma-radiation dose rates at given interior points of a light aircraft carrier to the dose rate 3.5 feet above the center of the Flight Deck which was assumed to be covered uniformly with Co^{60}. The results of the computations were compared with values determined by an experiment on the ship where the Flight Deck was uniformly contaminated with Co^{60}. The comparison was made for two portions of the ship which have different structural components. The results of the comparison showed the computed values to be higher than the experimental values and generally within 28% in the relatively "open" portion of the ship and generally worse than 50% in the more heavily compartmentalized portion of the ship.

125. Hanchon, K. B.

A method for computing a generalized effective fast neutron removal cross section. General Electric Co. Aircraft Nuclear Propulsion Dept., Evendale, Ohio.

In Paper 1 of PAPERS FROM SEVENTH

125. (cont'd) SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 6p.

An analytical method for computing a generalized effective neutron removal cross section is discussed. Such cross section would be a function of backing material and source-receiver distance, and would be specifically originated toward a slab geometry. Discussion of results obtained from the method are presented.

126. Hanchon, K. B. and Pope, M. L.

SHIELDING COMPUTER PROGRAM 19-0, FAST NEUTRON REMOVAL CROSS SECTION COMPUTATION (ANP PROGRAM NO. 514).

General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinatti. Rept. no. TID-11570; XDC-60-10-133, 4 Oct 60, 31p.

(Contracts AF33(600)-38062 and AT(11-1)-171)

Shielding Computer Program 19-0 computes fast-neutron effective-removal cross sections for those calculations of fast-neutron penetration that combine an influence function of the Albert-Welton form with an integration over a disk or cylindrical source. The cross sections thus calculated, may be considered as functions of input dose rates, material thickness and configuration, thickness of backing material, and source-receiver distance. Program 19-0 includes the following iterative capability; once the fast-neutron effective-removal cross section for a given material has been computed, that material may be regarded as a "backing material" and used in the computation of the fast-neutron effective-removal cross section of an inserted material.

127. Happe, R. A.

Materials in space. ORDNANCE v. 45, n. 244, p. 578-80, Jan-Feb 61.

The Ranger vehicle will subject metals and compounds to a new kind of environment in which performance can only be estimated. Materials to be used in the vehicle and some methods of evaluating their probable performance characteristics are described.

128. Hargrove, C. D. and Grum, A. F.

A STUDY OF THE ENGINEERING ASPECTS OF EXPEDIENT GAMMA-SHIELDING MATERIALS. Army Engineer Research and Development Lab., Fort Belvoir, Va. Rept. no. ERDL-1622-TR, 7 Apr 60, 72p. ASTIA AD-238 440

Tests were made to determine attenuation coefficients and dose rates as a function of shield thickness for the following shielding materials: water, limestone, sandstone, granite, sand, clay, silt, organic top soil (loam), fir, and oak. The following gamma emitters were used: Hg^{203}, Cs^{137}, Rb^{86}, Pr^{142}, and Na^{24}. Measurements were made with a NaI (T) crystal and a 128-channel pulse height analyzer. Point sources were used for the attenuation coefficient measurements and 12-in. diameter uniformly distributed sources for the dose rate measurements. Attenuation coefficients were plotted as a function of energy for the various materials. Dose rates are plotted as a function of shield thickness and gamma energy for the various shield conditions.

129. Harris, T. E.

WEIGHT-FEASIBILITY CALCULATION FOR SHIELDING OF TRUCK PASSENGERS. RAND Corp., Santa Monica, Calif. Rept. no. RM-1624, 3 Feb 56, 24p.

A shielded structure suitable for carrying passengers on a truck is proposed. The weight of shielding material required for a given degree of shielding is calculated on the basis of a roughly optimum distribution of the material. The results of the calculations are shown graphically. It is suggested that the results be used in accordance with the included modifications.

130. Harwood, J. J. (editor)

THE EFFECTS OF RADIATION ON MATERIALS. Reinhold Publishing Corp., New York, Copyright 1958.

130. (cont'd) Papers dealing with metals, alloys, inorganic dielectrics, semiconductors, organic and polymeric materials and materials for nuclear reactor components, including fuel elements, moderators, coolants and shielding materials are given.

131. Hashimoto, K., Kuroshawa, T. and Kaneko, M.

Experimental studies on shielding materials against ionizing radiations. REPTS. GOV. CHEM. IND. RESEARCH INST. TOKYO v. 55, p. 155-64, May 1960. (In Japanese)

The measurements were made on several samples of concrete made of portland or aluminous cement containing magnetite or barite as heavy aggregate for shielding γ rays, and those containing boron compound or hydrous aggregate for absorbing neutrons. The attenuation coefficient for Co^{60} γ rays was measured by a γ-ray spectrometer and the absorption coefficient was calculated. By the measurement of count as well as γ dose build-up, it was found that barite is better than magnetite as an aggregate. The oblique attenuation of γ rays was measured by a Geiger-Mueller counter, and it was confirmed that its build-up is usually larger than that of perpendicular incidence, but the crossover between the two may occur within a relaxation length. "Krillium" was tested to prevent the heavy aggregates from sinking in course of the setting of cement, and its effectiveness was recognized.

132. Helvey, T. C.

EFFECTS OF NUCLEAR RADIATION ON MEN AND MATERIALS. John F. Rider Publisher, Inc., New York, N. Y. Copyright 1959.

A general pamphlet reviewing the basic concepts of various types of radiation, radiation effects on men and materials, and shield configuration.

133. Hildebrand, R. H.

ELASTIC SCATTERING OF 83-Mev NEUTRONS. (Thesis) Univ. of Calif., Radiation Laboratory, Berkeley, Calif.

133. (cont'd) Rept. no. UCRL-1159, 5 Mar 51. (Contract W-7405-eng-48)

A relatively high intensity beam of 90 Mev neutrons from the 184-inch cyclotron was used to explore the elastic scattering patterns of Be, C, Al, Cu, Ag, and Pb. By measuring the ratio of the scattered neutron flux to the flux incident on the scattering nuclei it was possible to determine the absolute differential scattering cross sections. Total scattering cross sections were obtained by interaction of the differential cross sections and also by good and poor geometry attenuation experiments with Al, Cu, and Pb.

134. Hill, C.W.

Cylindrical shield Monte Carlo. Lockheed Aircraft Corp., Marietta, Georgia. In Paper 14 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER, 1959. TID-6302, 12p.

Monte Carlo calculations of single air-scattered γ transfer functions were performed using a cylindrical geometry. Center line dose was calculated for a typical case. An estimate of the variation of dose along the axis is shown. End effects were approximated by a simple cutoff.

135. Hill, C.W. and Ritchie, W.B.

NUCLEAR WEAPONS RADIATION DOSES IN ARMORED VEHICLES. Lockheed Nuclear Products, Marietta, Ga. Rept. no. NP-10038, p. 253-65.

A review is given of the methods used in analytic study for predicting the dose to crew members of an M-48 tank exposed to a nuclear weapons burst. The variation of armor thickness, 1 to 7 in., combined with the anisotropy of the radiation field, indicates that the dose is quite sensitive to burst orientation and detector position. The comprehensive calculations are made by mapping the vehicle surface with a complex polyhedron. Vehicle geometry parameters and dose transmission for the vehicle geometry are calculated once for each vehicle, while crew dose calculations are made for each source orientation and range.

136. Hine, G. J. and McCall, R. C.

Gamma-ray backscattering.

NUCLEONICS v. 12, n. 4, p. 27-30,

Apr 54.

Intensity and energy of γ rays backscattered from various materials are important in many shielding, absorption, and counting problems. This article describes the scattered γ-rays as functions of the energy of primary γ -radiation and geometry.

137. Hokkyo, N. and Kitazume, M.

Two group calculation of capture gamma-rays from infinite slab. (Hitachi Ltd., Tokyo). J. ATOMIC ENERGY SOC. JAPAN v. 2, p. 474-7, Aug 1960. (In Japanese)

The expression for the intensity of γ radiation that is generated in an infinite slab shield owing to thermal-neutron capture and escapes across the front and rear faces of the material is obtained for given fast- and thermal-neutron currents incident on the shield. The result is based on the two-group approximation for the thermal-neutron distribution and is a natural extension of Ilffe's result, which does not take the incident fast neutron into account.

138. Homer, E. N. and Naugle, J. E.

Radiation environment in space.

SCIENCE v. 132, n. 3438, p. 1465-72,

18 Nov 1960.

This article reviews the types and amounts of radiation in space, hazards in space travel, shielding requirements, and effects of space radiation to solar cells.

139. Horton, C. C.

PIPPA SHIELD DESIGN. Great Britain

Atomic Energy Research Establishment,

139. (cont'd) Harwell, Berks, England. Rept. no.

TRDC/P-3, 9 July 53, 13p.

Previous calculations on the Pippa shield are revised in the light of fresh design data for Pippa and recent experimental data. The shield thickness of concrete, heat production in the shield, the activity of the pressure vessel, the activity of coolant and void air, and the activity of the asbestos linings are considered. Thermal neutron currents are calculated for the radial case, and for the axial case the observed datum from Windscale is used: that the emergent axial current is 0.01 of the maximum thermal neutron flux at the center. Flux calculated by alternative methods are shown for comparison.

140. Horton, C. C.

Reactor shielding analysis. NUCLEAR

ENG. v. 3, n. 33, p. 515-20, Dec 1958.

Reviews the problems presented in the design of biological shielding, the general state of shielding analysis, and the way in which it differs from the core design analysis.

141. Horton, C. C.

The shielding of helical ducts. Rolls-

Royce, Ltd., Derby, England. NUCLEAR

SCI. AND ENG. v. 6, p. 525-9, Dec 1959.

The problem of radiation penetration through a shield containing a helical duct is considered by representing the duct by successive straight ducts joined at a fixed angle. Expressions for the emergent radiation flux are derived and illustrated for a typical case. It is shown that, in a representative example, the contribution from radiation scattered along the duct is a small fraction of the general increase in radiation level due to the reduced effective density of the bulk shield in the vicinity of the helix.

142. Hosoi, J., et al

Thermal properties and heating and

cooling durability of reactor shielding

concrete. Nihon Cement Co., Ltd.,

142. (cont'd) Research Lab. J. ATOMIC ENERGY SOC.

JAPAN v. 1, p. 308-18, 1959. (In Japanese)

A study was made of the thermal properties of various concretes made of domestic raw materials for radiation shields of a power reactor and of a high-flux research reactor. The results of measurements of thermal expansion coefficient, specific heat, thermal diffusivity, thermal conductivity, cyclical heating, and cooling durability are described. Relationships between thermal properties and durability are discussed and several photographs of the concretes are given. It is shown that the heating and cooling durability of such a concrete which has a large thermal expansion coefficient or a considerable difference between the thermal expansion of coarse aggregate and the one of cement mortar part or aggregates of lower strength is very poor. The decreasing rates of bending strength and dynamical modulus of elasticity and the residual elongation of the concrete tested show interesting relations with the modified thermal stress resistance factor containing a ratio of bending strength and thermal expansion coefficient. The thermal stress resistance factor seems to depend on the conditions of heat transfer on the surface and on heat release in the concrete.

143. Hubbell, J. H. and Spencer, L. V.

GAMMA -RAY ENERGY AND ANGULAR DISTRIBUTIONS IN CONNECTION WITH PENETRATION THROUGH THICK BARRIERS, PARTICULARLY LEAD. Wright Air Development Center, ANP Advisory Committee for Nuclear Measurements and Standards. Rept. no. WADC-TN-57-298, Pt. 1, Feb 58.

Qualitative features of gamma-ray spectra and angular distributions at deep penetrations are discussed on the basis of results of moments method and Monte Carlo calculations. Also, some aspects of spectroscopy as applied to verification of deep-penetration theory are brought out in the presentation of lead-penetration experiments.

144. Hullings, M. K.

THE EFFECT OF SOME LIQUID-METAL DUCTS ON REACTOR SHIELDS. Oak Ridge National Lab., Tenn. Rept. no. CF-53-11-53,

144. (cont'd) Feb 53. Decl. with deletions 12 Feb 57, 63p.

(Contract W-7405-eng-26)

A series of neutron attenuation experiments to test engineering mockups of liquid-metal ducts in reactor shields is reported. The experiments were performed in the thermal column of the ORNL Graphite Reactor for the purpose of acquiring sufficient data to facilitate calculations of additional shielding required by loss in efficiency in the reactor shield due to duct perforations.

145. Hungerford, H. E., Mantey, R. F. and Van Maele, L. P.

New shielding materials for high-temperature application. Atomic Power Development Associates, Inc., Detroit, Mich. NUCLEAR SCI. AND ENG. v. 6, p. 396-408, Nov 1959.

Investigation and development of several new materials for high-temperature shields have yielded three reasonably cheap materials which are structurally stable and able to withstand high temperatures and high radiation fields. Calculations indicate good neutron attenuation properties. These materials have undergone extensive development and testing for both physical and radiation effect data. They are (1) serpentine rock, (2) calcium borate, and (3) borated diatomaceous earth. Serpentine rock ($3\ MgO.\ SiO_2 \cdot 2H_2O$), as asbestos mineral, retains its water of hydration to temperatures as high as $950^{o}F$. It can be used either dry-packed, or as the aggregate in concrete, with densities attainable of about 130 lb/cu ft. Structurally, the aggregate is not quite as good as concrete. Calcium borate is the commercial name applied to a number of borated calcium minerals pressed into an asbestos matrix to give a boron content of about 12 wt.%, with a density of over 70 lb/cu ft. Although the composite is brittle, it can be fabricated into shapes rather easily. Tests indicate it will withstand temperatures up to $1800^{o}F$ with less than 3% shrinkage, and can be exposed to a neutron irradiation of 2.4×10^{20} nvt without damage. Diatomaceous earth, a porous commercial refractory material, was successfully borated to the extent of about 2 wt.% boron. It can be used as an aggregate in portland or lumnite concrete to give good strength properties and densities of 78 to 82 lb/cu ft.

146. Hungerford, H. E.

SHIELDING OF A FAST BREEDER REACTOR - I. THE PRIMARY SHIELD.

Atomic Power Development Associates, Inc., Detroit, Mich. Rept. no. AECU-3701, 1957, 19p.

146. (cont'd) The differing shielding requirements of a fast breeder reactor and a thermal reactor are discussed. The functions of a stainless steel layer inside the reactor vessel of a fast breeder are outlined: neutron reflector, gamma-ray absorbing thermal shield, and an inelastic neutron scatterer to protect the vessel walls against radiation damage. The use of borated graphite outside the vessel is discussed. The rotating plug shield is described. Shielding calculations and a primary shield design drawing are given for the Fermi Fast Breeder Reactor.

147. Hunter, E. T.

THE USE OF PLASTIC SHIELDS TO REDUCE PERMANENT NUCLEAR RADIATION DAMAGE. U. S. Army Signal Research and Development Laboratory, Fort Monmouth, New Jersey. Rept. no. USASRDL-TR-2123, 15 June 60. ASTIA AD-239 157

Permanent alteration in DC electrical characteristics of germanium, homogeneous base, pnp, power transistors was determined as a function of the thickness of epoxy-type shielding material surrounding the device. Exposures were made at the Godiva reactor. Results indicate that most of the permanent damage is caused by fast neutrons, with a smaller part of the damage caused by some other component of the radiation, and that the epoxy shields thermalize many of the fast neutrons, thereby reducing the permanent damage suffered by the transistors.

148. Irvine, T. F. and Cramer, K. R.

THERMAL ANALYSIS OF SPACE SUITS IN ORBIT. Wright Air Development Div., Life Support Systems Lab. Rept. no. WADD-TN-60-145, May 60.

149. Jager, T.

Concrete, the convenient reactor shielding. ATOMIC WORLD v. 9, p. 416-18, 438, Dec 1958.

Information is presented on the composition of concrete for reactor shielding. Discussions are included on: shielding principles, the amount of water in the mix,

149. (cont'd) increasing the density of concrete, high-density aggregates, thermal and chemical effects on concrete, the selection and kinds of aggregates, the addition of boron compounds to the mix, and cost factors.

150. Jaeger, T.

Concrete in radiation-shield technique.

ATOMKERNEERGIE v. 2, n. 6, p. 217-22,

June 1957. (In German)

It is becoming increasingly necessary for concrete radiation shields to be design in detail. Therefore, the following important design topics are discussed: the neutron cross-sections of the various elements in the concrete; the mean free paths and relaxation times; the interaction of photons with shield materials, i.e. photoelectric effect, pair production and Compton effect.

151. Jaeger, T.

Concrete in radiation-shielding technique.

ATOMKERNENERGIE v. 2, n. 7, p. 255-62,

July 1957. (In German)

Experiments on the absorption of narrow and broad beams of radiation in ordinary and heavy concrete are described. From the experimental results, the mass absorption coefficient is evaluated as a function of gamma-ray energy and the increase of secondary x-rays as a function of beam width. The capture of neutrons, their elastic and inelastic scattering by various elements in concrete, are reviewed. Various simple shield geometries are discussed.

152. Jaeger, T.

Concrete in radiation-shielding technique.

ATOMKERNENERGIE v. 2, n. 8-9, p. 338-

44, Aug/Sep 1957. (In German)

Various methods for calculating the total attenuation of photo-radiation and the total effect of scattered radiation in a concrete shield are discussed. The neutron attenuation is calculated by the single-and two-group diffusion theory. As an example, the calculated and experimental results for the shielding of the Windscale reactors are shown.

153. Jaeger, T.

Concrete in radiation-shielding technique.

ATOMKERNENERGIE v. 3, n. 2, p. 69-72,

Feb 1958. (In German)

The temperature variation of stopping power and the alteration of the material by radiation is considered.

154. Jaeger, T.

Concrete in radiation-shielding technique.

ATOMKERNENERGIE v. 3, n. 3, p. 101-8,

Mar 1958. (In German)

The various physical properties of concrete that are important in shielding, such as high density, high water content, composition, etc., are surveyed for various cements and additives. The composition and cost of the available additives are reviewed. Finally, concreting techniques, e.g., "puddling" and the Intrusion-Prepakt method are discussed.

155. Jaeger, T.

OUTLINE OF RADIATION SHIELDING

TECHNIQUES FOR STRUCTURAL ENGIN-

EERS, PROCESS TECHNICIANS, HEALTH

ENGINEERS, AND PHYSICISTS. (Grundzüge

der strahlenschutztechnik für bauingenieure

verfahrenstechniker, gesundheitsingenieure,

physiker.) Berlin, Springer-Verlag, 1960,

406p.

Radiation shielding is outlined for the engineer. Section titles are: Atomic Physics Fundamentals, Radiation Detection Instruments, Radiobiology Fundamentals, Gamma and Neutron Sources, Geometry of Sources, Experimental Arrangements for Reactor Shielding Measurements, Calculation of Gamma Attenuation, Calculation of Neutron Attenuation, Heat Generation by Radiation, Thermal Shielding of Reactors, Biological

155. (cont'd) Shielding of Reactors, Design of Radioisotope Laboratories, Design of Separation Plants, Design of Technical and Medical Gamma Irradiation Facilities, Shielding of Particle Accelerators, Disposal of Radioactive Waste Materials from Nuclear Research and Nuclear Industry, Reactor Accidents and Their Consequences, and Safety Enclosures for Reactor Systems.

156. Kalos, M. H.

GAMMA RAY PENETRATION IN COMPOSITE SLABS. Nuclear Development Corp. of America, White Plains, New York. Rept. no. NDA-2056-10, 15 Mar 57. Decl. 10 June 60, 56p. (For General Electric Co. Aircraft Nuclear Propulsion Dept. Subcontract AT-29)

Monte Carlo calculations were made to determine the penetrations of γ rays through composite slabs of matter. In the first series of calculations the build-up at normal incidence through three mean free paths of one material followed by varying amounts of another was determined. The results were compared with single-layer results. For lead and water combinations, rough prescriptions are given for the synthesis of double-layer results from the well-known single-layer build-up factors. In addition, calculations of the slant penetration of γ rays through composite slabs of polyethylene and lead were carried out and the results compared with those in the Reactor Handbook. The new results are in fair agreement with the old. The results were also compared, in part, with normal incidence rays to determine the role of "short-circuiting" of γ rays. It is concluded that within the range of this study the latter is not important for angles of incidence less than about 60°.

157. Kam, F. B. K. and Stern, H. E.

DEVELOPMENT OF AN IBM 704 ANALYTICAL CODE FOR ANALYSIS OF AXIALLY SYMMETRIC REACTOR SHIELDS. Oak Ridge National Lab., Tenn. Rept. no. ORNL-3016, p. 231-2.

Analytical IBM 704 codes, intended to furnish neutron and γ data applicable to pre-analysis of shielding experiments and to general shield design, were written. Fast neutron and primary γ doses outside of an axially symmetric reactor shield can be computed for detector points located along the axis of symmetry. Two subprograms are ready for use; the remaining codes are completed and are being debugged.

158. Kazanskiĭ, Yu. A.

The angular and energy distribution of γ radiation in water and iron. ATOMNAYA ENERG. v. 8, p. 432-4, May 1960. (In Russian)

Multiple scattering characteristics and angular and energy distributions were studied in order to evaluate γ radiation attenuation in complex geometries of iron and water. The angular and energy distributions from a Co^{60} source in water and in iron were measured under semi-indefinite conditions, and it is shown that the distribution maximum is found near the angle of minimum single scattering. The intensity distributions are of an exponential character; moreover, the exponential coefficient is a linear function of the atomic number of the medium.

159. Keller, J. W. and Schaeffer, N. M.

RADIATION SHIELDING FOR SPACE VEHICLES. Convair, Fort Worth, Tex. (Presented at 1960 Pacific General Meeting American Institute of Electrical Engineers, San Diego, California 8-12 August 1960). 15 June 1960, 29p.

Ramifications on shielding of manned space vehicles are being investigated in view of the existence of the intense radiation environment in space. The space radiation environment is reviewed, and the problem of selecting best shielding materials in view of this environment is treated. The results of preliminary calculations to determine requirements for shielding against Van Allen radiation and solar protons are given. These results indicate that for most missions (outside the heart of the inner Van Allen belt) exposure to solar protons will be the controlling factor in determination of shield weight, suggesting the possible use of two crew compartments, one for normal operations and a smaller heavily shielded one for occupancy following solar flares.

160. Keller, J. W.

THE SHIELDING OF SPACE VEHICLES.

George C. Marshall Space Flight Center,

160. (cont'd) Huntsville, Alabama. Rept. no. MTP-M-RP-61-12, 16 May 61.

The biologically important types of radiation fields to be encountered by the astronaut are reviewed briefly and the problems encountered in shielding against each are discussed. An effort is made to point out large uncertainties in the in-put data and shield calculations, and to indicate areas where refinements are most immediately needed.

161. Keller, J. W. and Schaeffer, N. M.

SHIELDING OF MANNED VEHICLES FROM SPACE RADIATIONS. Convair, Fort Worth, Tex. (Presented at the 31st Annual Meeting Aerospace Medical Association, Miami Beach, Florida, 9-10 May 1960). Rept. no. FZM-1979, 11 May 60.

Ramifications on shielding of manned space vehicles are being investigated in view of the existence of the intense radiation environment in space. In this paper, available data on radiation environment are reviewed and the problem of selection of best shielding materials on the basis of these data is treated. The results of preliminary calculations to determine the effectiveness of promising materials as shields against the protons and electrons in the Van Allen belts are given and the resulting shield requirements are discussed.

162. Keller, J. W.

A STUDY OF SHIELDING REQUIREMENTS FOR MANNED SPACE MISSIONS. Convair, Fort Worth, Texas. Rept. no. FZK-124, 10 Oct 60.

In this report the radiation environment is reviewed and the problem of selecting shield materials in view of this environment is treated. The results of preliminary calculations to determine requirements for shielding against Van Allen radiation and solar protons are given. These results indicate that for most missions (outside the heart of the inner Van Allen belt) exposure to solar protons will be the controlling factor in determination of shield weight, suggesting the possible use of two crew compartments - one for normal operations and a smaller, heavily shielded one for short-term occupancy following solar flares.

163. Kereiakes, J. G. and Krebs, A. T.

Further studies on the effect of partial shielding by grids on survival of x-irradiated rats. Army Medical Research Lab., Fort Knox, Ky. BRIT. J. RADIOL. v. 32, p. 339-41, May 1959.

Exposure of rats to the same mid-line doses of x-irradiation through grids of various ratios of open area to closed area but a constant diameter hole size resulted in significantly greater survival only for small ratios of open area to closed area. These results are in contrast to the findings found for selective shielding-type experiments, where the effect depends on the sensitivities of the shielded portions.

164. Chain-like packing material for sealing against radioactive radiation. GERMAN PATENT 1,057,268. KERNTECHNIK v. 2, p. 359, Nov 1960. (In German)

A chain-like plastic packing material for sealing against radioactive radiation is described. The material consists of metallic particles (lead, bismuth, or cadmium) of a size less than 12 μ (80 screen meshes per cm^2) which are coated with a thin film of binder. The mixture is then pressed in an extrusion press under a pressure of 35 atm, so that the density is 9. The patent application contains further theoretical metallographic considerations on the occurrence of this effect. The following were used as typical mixtures: 15 wt.% of a compressor lubricant (containing 72% heavy motor oil, ~20% stearic acid, aluminum, and a mineral filler) and 85 wt.% lead powder or 10 wt.% of a mixture of light and heavy lubricants with medium viscosity and 90 wt.% lead powder. Other metal powders can be used in place of the lead powder. Any oil that forms a coating film on metal particles can be used. The choice of the components depends on the type and conditions of application, the space to be filled, and the temperature. For the sealing of fissures, rents, or leaks in the presence of active radiation or outlets, such a material, especially with lead or bismuth, would prove very good. The metal particles have, in their passage through the extrusion press, the inclination to aggregate on the outer surface of the rod-like moldings so that a type of flexible metal strand is obtained which can be shaped easily to be worked in fissures or groves.

165. Keshishian, V.

SHIELD DESIGN FOR SNAP REACTORS.

Atomics International, Canoga Park, Calif.

(Presented at the ARS Space Power Systems Conference, Santa Monica, Calif., 27-30 September 1960). (American Rocket Society Inc., New York, N.Y., 1334-60)

The radiation originating from a space nuclear auxiliary power system of the SNAP type may be reduced to acceptable payload protection limits by proper vehicle, component, and shielding arrangement. Studies have shown that, depending on this arrangement, shield weights can vary from zero to a few thousand pounds. The high degree of interdependence between the shield and the vehicle is pointed out, and nine possible configurations with resulting shield weights are presented.

166. Kimel, L.R.

Build-up factors for heterogeneous shields. ATOMNAYA ENERG. v. 10, p. 173-5, Feb 1961. (In Russian)

The build-up of Co^{60} γ-radiation dose factors in heterogeneous (two-layer) shielding was determined for a parallel-plane beam with normal incidence to the shielding, consisting of lead, iron, and aluminum plates (75 x 75 cm.) The build-up factors were determined for combinations of Pb + Al, Al + Pb, Pb + Fe, Fe + Pb, Fe + Al, and Al + Fe, with the first material in the combination the closest to the irradiation source.

167. Kimel, L.R.

On the optimum shapes of shielding elements. ATOMNAYA ENERG. v. 7, p. 265-6, Sep 1959. (In Russian)

A geometric scheme is plotted for determining the shape of an optimum weight shielding element for an evenly distributed linear source. The derived equation $r = f(\varphi)$ describes the shape of the shielding barrier. An analogous procedure was used

167. (cont'd) for disc shaped sources. Dosage variations at the radiation scattering point were not considered in the calculations.

168. King, R. W.
U. S. NAVY REQUIREMENTS FOR SHIELDING INFORMATION. Naval Radiological Defense Lab., San Francisco, Calif. Rept. no. NP-10038, p. 25-37.

Shielding is a central countermeasure against radiation whether from controlled nuclear processes or phenomena resulting from nuclear weapons detonations. This fact dictates a military interest in knowledge of shielding capability and analytical techniques for its determination. The Navy has, therefore, supported shielding research in both its fundamental and applied aspects. The Navy's interest in engineering applications is summarized. Although Navy interest extends to sea, air, space, and land environments, application of shielding information in ships is emphasized.

169. Kittel, J. H.
Damaging effects of radiation on solid reactor materials. NUCLEONICS v. 14, n. 9, p. 63-65, Sep 1956.

This article covers important effects that have been found to affect the properties of moderators, structural materials, control materials, and shielding materials.

170. Klahr, C. N. and Held, K.
Nuclear shielding for space environment--the scattered shield. TRG, Inc., Syosset, N. Y. In Paper 15 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 14p.

170. (cont'd) Shielding of a nuclear reactor in a space vehicle should be designed to eliminate radiation, predominantly by scattering it into space rather than by absorption. This can eliminate much of the multiply scattered radiation (build-up) that would normally contribute to the dose. The following general rules for nuclear shadow shielding in space environment constitute what may be called the space scattering principles: (1) multiple splitting of the shadow shield to increase the probability of scattering into space; (2) shaping of each shadow disk to increase scattering into space; (3) reliance on good scatters with more isotropic cross section; and (4) the possibility of using low density materials without weight penalty because of the one dimensional geometry. Some schematic calculation results are presented.

171. Klahr, C. N.

Scattering shields for space power.

Technical Research Group, Inc., Syosset,

N. Y. NUCLEONICS v. 19, n. 4, p. 110, 111,

Apr 1961.

Rocket reactor shielding techniques are examined, including the use of multiple shields, the shaping of shields, and the selective positioning of these shields to greatest advantage. Weight, heating effects, and configuration problems concerning shielding are discussed. Secondary radiation effects are noted.

172. Kleinecke, D. C.

THE EFFECT OF AN AIR-SAND INTERFACE ON GAMMA-RAY TRANSPORT.

Univ. of Calif., Berkeley. Rept. no.

NP-10038, p. 301-7.

A description is given of Monte Carlo calculations for gamma flux from an isotropic, monoenergetic point source lying on an interface between air and sand. The results include the sample means and standard errors tabulated separately for each order of scattering for the energies considered: 0.097, 0.280, 0.662, and 1.700 Mev.

173. Klimushev, A. V. and Merkulov, V. S.

A criterion for the selection of shielding

diaphragms for measurements by the method

173. (cont'd) of attenuation of beta radiation. PRIBOROSTROENIE n. 10, p. 30-1, 1960.

When using the method of attenuation of beta radiation for measuring the density of gases, it is frequently necessary to use shielding diaphragms. As the accuracy of measurement is reduced by these diaphragms, it is necessary to reduce their thickness. A corresponding decrease of the radiation flux leads to an increase of the statistical error in the recording of radiation. These factors are considered for an optimum selection of the dimensions of shielding diaphragms.

174. Kloster, R. L.

ANALYSIS OF THE MOMENTS METHOD EXPERIMENT. Convair, Fort Worth, Tex. Rept. no. NARF-59-33T; FZK-9-138; Project 61(1-9964), 4 Sep 59, 35p.

Monte Carlo calculations show the effects of a plane water-air boundary on both fast neutron and gamma dose rates. Multigroup diffusion theory calculation for a reactor source shows the effects of a plane water-air boundary on thermal neutron dose rate. The results of Monte Carlo and multigroup calculations are compared with experimental values. The predicted boundary effect for fast neutrons of 7.3% agrees within 16% with the measured effect of 6.3%. The gamma detector did not measure a boundary effect because it lacked sensitivity at low energies. However, the effect predicted for gamma rays of 5 to 10% is as large as that for neutrons. An estimate of the boundary effect for thermal neutrons from a PoBe source is obtained from the results of multigroup diffusion theory calculations for a reactor source. The calculated boundary effect agrees within 13% with the measured values.

175. Knapp Mills, Inc.

METHOD FOR MAKING LEAD SHIELDS. BRITISH PATENT 862,571, 15 Mar 61.

A method is given for making lead and lead alloy shields of uniform density and free from voids, shrink cavities and similar defects. In this method, a layer of molten lead is placed in an enclosure and cooled from underneath to solidify the bottom progressively, the top surface being kept molten by heating with torches. The condition of the surface of the solidified lead covered by molten lead is determined with probes, and more molten lead can be added at intervals to give the desired depth. In this way, it is possible to make lead shields of any thickness without fear of defects or cavities. Enclosures for making various lead shapes, including one for a container, are described.

175A. Kodyukov, V. M.

Attenuation of the γ-rays from point sources by various mediums. ATOMNAYA ENERG. v. 9, p. 140, Aug 1960. (In Russian)

The attenuation of γ rays from 1-c sources of Au^{198} (E = 0.411 Mev), Zn^{65} (E = 1.12 Mev), and Na^{24} (E_1 = 1.38 Mev and E_2 = 2.76 Mev) in water, lead, and iron was investigated. The tabulated data show that the build-up factor ($V = K_{theor.}/K_{exp.}$) is a function of boundary conditions with an increase in absorber atomic number and γ energy the effects of the medium boundary are reduced.

176. Komarovskii, A. N.

The economics of heavy concrete as a radiation shield. ATOMNAYA ENERG. v. 4, p. 437-42, May 1958. (In Russian)

A review is given of foreign publications on the use of various types of concrete for biological protection against radiation from nuclear reactors and heavy particle accelerators. The economics of heavy concrete shields or ordinary concrete with mineral admixtures are discussed. Calculations and economic analysis made by Russian specialists are included. The results show that special heavy concrete shielding should be used only in exceptional cases.

177. Komarovskii, A. N.

Evaluation of the economic expediency in using special, heavy concrete for the shielding of radiation. (Otsenka ekonomicheskoi tselesoobraznosti primeneniya spetsial'nykh tiazhelykh betonov dlia zashchity of izlucheniya). ATOMNAYA ENERG. v. 4, p. 437-42, 1958. (Translated by L. Venters, Argonne National Lab.)

An evaluation was made of the economic expediency in using special, heavy concrete or ordinary concretes with mineral fillers for a biological shield for nuclear reactors

177. (cont'd) and accelerators of charged particles. A survey was made of the literature. Comparative data on various types of concrete shielding are tabulated. Calculations on shielding efficiency are included. Based on the data collected it was concluded that the use of special, heavy concretes is expedient only in exceptional cases.

178. Komarovskii, A. N.

Heating of the structures around a nuclear reactor. ATOMNAYA ENERG. v. 5, p. 119-23, 1958. (In Russian)

The effects of heating and radioactive emission from nuclear reactors on the enclosing concrete shielding are analyzed. Descriptions are given of various types of concretes. Data are presented on the mechanical properties and moisture content of concrete shielding exposed to heating and irradiation in operating reactors. Data are also given on materials used for reactor thermal shielding and methods of cooling the shield.

179. Komarovskii, A. N.

Shielding materials for nuclear reactors. (Translated from the Russian by V. M. Newton. H. W. Curtis, translation ed. International Series of Monographs on Nuclear Energy. Div. VII.) REACTOR ENGINEERING v. 1. New York, Pergamon Press, 1961, 151p.

Concretes used in reactors and accelerators are examined; shielding, technological, and engineering properties of these concretes are studied. Particular attention is given to special heavy and hydraulic concretes. Radiation hazards, construction methods, and economic considerations are summarized for the concretes studied.

180. Kovalev, E. E. and Osanov, P. D.

Influence of the size of an extended

plane source on gamma-ray attenuation

in shielding materials. REACTOR TECHNOL.

v. 1, p. 52-4, Apr 1959.

A treatment of the effects of source dimension on gamma attenuation is presented. Data are given to illustrate the conclusion that the large dimensions of an extended source influence the estimated gamma attenuation in two ways: the existence of oblique rays increases the calculated attenuation and multiple scattering decreases it more strongly than would be calculated. Thus, corrections can be made for these effects in shielding calculations.

181. Krauss, G. V.

S3G/S4G PRIMARY SHIELD WATER

SYSTEM; DESIGN DESCRIPTIONS.

Knolls Atomic Power Lab., Schenectady,

N. Y. Rept. no. KAPL-M-SSD-43, 27 Sep

57, 18p. (Contract W-31-109-eng-52)

The function of the Shield Water System is to provide shielding to protect personnel from excessive neutron radiation. Shielding is accomplished by enclosing the pressure vessel sides and bottom in an annular shaped tank which is maintained full of water at atmospheric pressure. The Shield Water System consists of the neutron shield tank which surround the pressure vessel, the shield water expansion tank, interconnecting piping and valves, instruments and an alarm system. The expansion tank insures that the shield tank will be full of water at all times by maintaining a supply of water in reserve and by providing adequate facilities for adding additional water while the reactor plant is in operation. The expansion tank also provides for volume changes due to thermal expansion of the shield water. The interconnecting piping and valves are used for filling, adding rust inhibitor, and draining. In the event of low level in the expansion tank, an alarm located on the RPCB and an alarm light located in the vicinity of the expansion tank will be energized warning the operator of this condition.

182. Kereiakes, J. G. and Krebs, A. T.

GRID VARIABLES INFLUENCING THE SURVIVAL OF X-IRRADIATED MICE. Army Medical Research Lab., Fort Knox, Ky. Rept. no. AMRL-403; Project no. 6-59-08-014-01, 9 Nov 59, 10p.

The studies on partial-body shielding by grids confirm the beneficial effects of this radiation dose distribution procedure on the survival of x-irradiated mice. This beneficial influence is exerted only if the dose transmission to the tissue surrounding the damaged tissue does not exceed 8 to 10% (for 900 r midline tissue dose.) In the otherwise supralethal dose range, shielding by the use of grids does not result in a significant increase in survival. At a 1200 r midline dose, the survival time is still significantly greater for smaller aperture size grid. However, at midline doses higher than 1600 r, mice irradiated through smaller aperture size grid have a shorter mean survival time. Radiation dose transmission through closed area of grid has greater effect on smaller aperture size grid.

183. Ksanda, C. F.

SHIP SHIELDING CALCULATIONS. Naval Radiological Defense Lab., San Francisco, Calif. Rept. no. NP-10264, p. 135-59.

The general approach to computing ship shielding factors is outlined. In this approach, the dynamic source configurations produced by various nuclear detonations, the complex structures of ships, and interactions of radiations with ships are idealized. Equations for calculation purposes are discussed. Shielding factors derived from Test Baker of Operation Crossroads were found to give good agreement with calculated values. A pseudospectrum for iron and air or water is presented.

184. Ksanda, C. F.

SHIP SHIELDING CALCULATIONS COMPUTATIONAL RESULTS. Naval Radiological Defense Lab., San Francisco, Calif.

184. (cont'd) Rept. no. NP-10264, p. 59A-159N.

Shielding calculations for a number of important locations within the USS RANGER were completed for two cases: (1) activity deposited on the flight deck and (2) activity air-borne around the ship. The results for both cases are presented as plots of shielding factor vs. total plating thickness directly above the receiving point.

185. Kukhtevich, V. I. and Tsypin, S. G.

Attenuation of the neutron-capture γ-rays created in iron-water shields. REACTOR TECHNOL. v. 1, p. 54-5, Apr 1959.

The relaxation length of neutron-capture gamma rays in iron-water shields was measured as a function of the proportion of Fe. Neutrons of 4.0 and 14.9 Mev were used.

186. Kukhtevich, V. I. and Tsypin, S. G.

Attenuation of gamma-rays from neutron capture in iron-water mixtures. KERNENERGIE v. 1, p. 385-6, May 1958. (In German)

The effects of Fe concentration in H_2O on the relaxation length of γ rays from neutron capture were studied. Neutrons from D(d,n) and T (d,n) reactions were used. The arrangement used was a tank with 60 cm^2 Fe plates with water spaces between. A 1.5-cm^3 ionization chamber detected the γ rays. Curves for 4-Mev and 14.9 Mev neutrons are given which both show a minimum γ relaxation length at 60 vol.% Fe.

187. Lane, J. A.

FISSION PRODUCT ACTIVITY OF X SLUGS. Oak Ridge National Lab., Tenn. Rept. no. CF-50-5-29, 2 May 50. Decl. 15 Feb 57, 6p. (Contract W-7405-eng-26)

Fission product activity measurements of MTR slugs were made for design data for discharge chute shielding and channel water depth.

188. Lavie, J. M.

Slide rule of shielding against gamma radiation (Centre d'Études Nucleaires, Saclay, France) BULL. INFORM. SCI. ET TECH (PARIS) n. 39, p. 43-8, Apr 1960. (In French)

A simple slide rule permitting the rapid solution with a good approximation of all the practical problems connected with the prevention and protection against external gamma irradiation is presented. The slide rule was designed for a point source, but its range of utilization can be extended to a surface source with a good approximation.

189. Lawson, B. G., Billings, M. S. and Bennett, L. R.

LATE EFFECTS OF TOTAL-BODY ROENTGEN IRRADIATION. LONGEVITY AND INCIDENCE OF NEPHROSCLEROSIS AS INFLUENCED BY PARTIAL-BODY SHIELDING. Univ. of California, Los Angeles. Atomic Energy Project. Rept. no. AF-SAM-59-33, 29 Sep 58, 15p.

Two hundred forty-two female Wistar rats were observed throughout their life span following 1,000 r hypoxic total-body or partial-body irradiation. Hypoxid radiation with superimposed anesthesia resulted in 67 percent 30 day mortality, compared to 9 percent mortality without anesthesia. Selection of the colony by acute postirradiation deaths did not influence the magnitude of late radiation sequelae as measured by life-shortening. Growth retardation during the second postirradiation year was well correlated with life-shortening. Life-shortening was observed after partial-body irradiation to an extent approximately proportional to the weight of radiated tissue. Nephrosclerosis was not observed unless the upper abdomen was included within the radiation field. Other than nephrosclerosis, a similar incidence of disease was observed at death in control and irradiated rats whether partial-body or total-body irradiated.

190. Ledoux, J. C.

ANALYSIS OF THE CRITICAL SHIELDING VOLUME FOR UNDERGROUND SHELTERS.

Naval Civil Engineering Lab. Port Hueneme, Calif. n.d., 30p. ASTIA AD-250 640

The development of design principles for constructing atomic warfare shelters for naval shore establishments is described. An investigation was made to determine which part of the earth covering a buried shelter is the most important as a radiation shield. The equation $(1-F)\ E_1(u_1x) = E_1(u_1\ x\ \sec\ \theta)$, when solves for the critical angle, θ will define the volume of earth which provides the fraction F, of the total shielding to the shelter system. E is the exponential integral, u_1 is the effective linear absorption coefficient for the shielded material, and x is the effective shield thickness. Computations were completed for slab and hemisphere geometry, x was equivalent to t_m, the thickness of the slab shield. For hemispherical geometry, was equivalent to R_m which was the sum of the radius of the hemisphere plus the minimum cover over the arc. In order to preserve the shielding integrity of the shelter system, the shielding volume defined by θ should not be violated by openings of any sort.

191. LeVine, J. H.

RADIATION EFFECTS ANALYSIS METHODS FOR NUCLEAR AIRCRAFT DESIGN. (Paper presented at the First Semi-Annual 125A Radiation Effects Symposium, 22-23 May 1957 sponsored by the U. S. Air Force, at Convair, Fort Worth, Texas).

The advent of a nuclear-powered aircraft utilizing the divided shield concept necessitates the evaluation of aircraft parts and systems in a nuclear environment. Functional thresholds for aircraft parts and systems may be obtained utilizing the "weak link" concept, and an ultimate system life computed in a known nuclear environment. A method for building radiation resistance into a system is discussed.

192. Levinger, J. S.

Can the nuclear many-body problem be solved by using perturbation theory?

Louisiana State Univ., Baton Rouge, La.

NUCLEAR PHYS. v. 19, p. 370-6, 1 Nov 1960. (In English)

A static two-body potential of Serber exchange character with a repulsive core failed to produce saturation in first-order perturbation theory, if the core was treated as a pseudo-potential. (The order of the term was designated in a joint expansion in powers of the strength of the attractive potential and in the range of the repulsive core.) Second-order terms were estimated from the condition that saturation should be achieved at the observed density without invoking many-body forces. It was concluded that any static potential that produces saturation at about the empirical density must have substantial second-order terms of magnitude 10 Mev/particle or more. The use of a velocity-dependent two-body potential to replace the infinite repulsive core was discussed. It was found that second-order terms may be much smaller if a velocity-dependent potential is used.

193. Levy, R. H.

RADIATION SHIELDING OF SPACE VEHICLES BY MEANS OF SUPER-CONDUCTING COILS. Avco-Everett Research Laboratory, Everett, Mass.

Rept. no. 106; AFBSD-TN-61-7, Apr 61, 37p. (Contract AF 04(647)-278)

The general problem of shielding the occupants of manned space vehicles from various radiations likely to be encountered in space flight is discussed and various published papers on the subject are briefly reviewed. The review indicates the importantance of the problem and the interest that would attach to a radical solution. One possibility is shielding by the permanent magnetic field of a superconducting coil. A detailed analysis is made of the shielding that could be provided by such a coil and a preliminary estimate of the weight of such a device is made paying particular attention to the weight of the structure required to support the coil. A comparison is made of the weights calculated in this way with the weight of the spherical H_20 shield which would give comparable protection.

194. Lindackers, K. H.

Shielding calculations for nuclear reactors.

Part II. (Technische Überwachungs-Vereins,

Cologne). ATOMKERNENERGIE v. 5,

p. 379-83, Oct 1960. (In German)

In this last section of the report on shielding calculations for reactors, the calculation of the effect of γ radiation on the dose output at the reference point is described. It is concluded that the total RBW dose output at a reference point is the sum of the RBW dose output calculated for the individual energy groups of the neutrons and γ radiation.

195. Lockheed Aircraft Corp., Missile Systems Div.,

Sunnyvale, California

SHIELDING AGAINST FAST PROTONS

IN THE TRAPPED RADIATION BELT.

Rept. no. LMSD-48395, 15 Jan 59.

A discussion of the present state of knowledge on the trapped radiation belt, radiation tolerances of possible payload components, typical shield designs and weights, suggested modifications and improvements of the present calculations and a bibliography of experimental and theoretical articles on proton energy loss, straggling, and secondary production.

196. Lockyer, I. J.

SHIELDS FOR NUCLEAR REACTORS.

Morgan Crucible Co., Ltd. BRITISH

PATENT 821,915, 14 Oct 59

The preparation of a carbon-graphite-boron shielding material is described. In one example 50 parts graphite dust, 25 parts petroleum coke, 20 parts borax, and 30 parts tar or pitch binder are combined, tamped, and heat treated at 1000°C. The capture cross section is 2.52 cm^2. Alternatively, a petroleum coke and binder mix is tamped and heat treated at 1200°C, and then dipped in molten borax under air pressure to produce a material with a capture cross section of 0.477 cm^2.

197. Long, E. and Jones, J.

IMPROVEMENTS IN OR RELATING TO

SHIELDING FOR NUCLEAR REACTORS.

United Kingdom Atomic Energy Authority.

BRITISH PATENT 836-147, 1 June 60.

The design and materials for the neutron shield of a reactor are reported. The neutron shield is placed adjacent to the core of the reactor. The shield has holes passing through it to provide access to the core. Tubes are arranged in the holes and end in neutron shield plugs. The materials used in the construction of the shielding and tubes have a high neutron absorption cross section.

198. MacDonald, J. E. and Trampus, A.

A Monte Carlo program for air-scattered

neutrons. Aircraft Nuclear Propulsion

Dept., Evendale, Ohio. In Paper 2 of

PAPERS FROM SEVENTH SEMIANNUAL

SHIELDING INFORMATION MEETING,

14-15 OCTOBER 1959. TID-6302, 7p.

A description is given of the application of the Monte Carlo method in the calculation of the energy and angular distribution of neutrons scattered in air. The events considered are: isotropic elastic scattering, anisotropic elastic scattering, inelastic scattering, radiative capture, and absorption without subsequent particle emission. The probability density and scoring functions are exhibited which will make up the expected values that are to be estimated by random sampling techniques. These basic equations are shown to yield to such variance reducing methods as statistical estimation, importance sampling, and quota sampling.

199. MacDonald, J. E. and Martin, J. M.

SHIELDING COMPUTER PROGRAM 20-0.

General Electric Co. Aircraft Nuclear

Propulsion Dept., Cincinnati. Rept. no.

199. (cont'd) TID-11910; DC-60-10-98, 21 Oct 60.

Program 20-0 generates and writes on tape source particle parameters to be used as input for Monte Carlo shield programs of the 18 series designed for analysis of reactor-shield assemblies. Source particles are generated in source tubes defined by right circular cylinders with mutually parallel axes of symmetry. A special case arises when the entire reactor is treated as a single source tube. The case of a point source can also be handled. Source particle spatial coordinates are chosen from appropriate power and power density distributions by one of two methods; a random method based on uniformly distributed random numbers, and a systematic method that determines the number of source particles to be started from specified volume elements. The systematic method is the only method that can be used for the special case of a single source tube. The energy of each source particle is chosen by a random method from an energy spectral distribution that is space independent. Provision is made in the program for splitting on region and energy to conform to the demands of the 18 series programs. Program 20-0 is coded for use on an IBM 704 having 32,768 magnetic core memory locations. Five magnetic tape units are used by the program.

200. MacDonald, J. E. Martin, J. T. and Yalch, J. P. SPECIALIZED REACTOR-SHIELD MONTE CARLO PROGRAM 18-0. Aircraft Nuclear Propulsion Dept., General Electric Co., Cincinatti. Rept. no. TID-11681; XDC-61-9-91, 13 Jan 61, 117p. (Contracts AF33(600)-38062 and AT(11-1)-171)

Specialized Monte Carlo Program 18-0, GE-ANPD Program 349, is a digital computer program that applies Monte Carlo methods to simulate neutron and γ-ray life histories in reactor-shield assemblies. The code is specialized to provide only as much geometry and importance sampling capabilities as are needed for analysis of selected reactor-shield systems. Specialization in these areas results in a more efficient code for analysis of reactor-shield systems than can be achieved by application of more general-purpose Monte Carlo codes. The program is coded for an IBM-704 with a fast memory capacity of 32,768 storage locations. Seven magnetic tape units are required, but no magnetic drum storage is necessary.

201. Maienschein, F. C., Peele, R. W. and Love, T. A.

Recent fission gamma-ray spectral measurements. Oak Ridge National Lab., Tenn.

In Paper 3 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 13p.

Measurements of the γ-ray spectrum closely associated in time with the fission of U^{235} by thermal neutrons were extended down to 10 kev with a single-crystal spectrometer. Preliminary results indicate a photon yield which varies between 6 and 15 photons/fission/Mev for energies from 10 to approximately 600 kev except at the heavy fission-fragment x-ray peak where the yield increases to about 24 photons/fission/Mev. In another investigation, an unrefined measurement of the gross spectrum of γ rays associated with the thermal fission of U^{235} was made with a three-crystal pair spectrometer and an irradiated U^{235} foil. Within the limitations, the results roughly agree in spectral shape with the data of Motz and with a summation of the earlier prompt-fission and fission-product data.

202. Malamud, E.

Absorption of 1-Bev photons. Cornell Univ., Ithaca, N. Y. PHYS. REV. v. 115, p. 687-94, 1 Aug 1959.

The total cross section for the attenuation of high-energy gamma rays was measured in various elements. The variation with atomic number, Z, was observed by measuring the absorption of 1-Bev gamma rays in 12 different elements ranging from hydrogen to uranium. Additional measurements were made in copper at 400 and 700 Mev to show the energy dependence of the absorption processes. These results are then combined with those of other investigators at lower energies and show that the theory of pair production in the nuclear field correctly predicts the cross section as a function of atomic number and photon energy. The measurements in low-Z elements give information on pair production in the field of the electron. The results are in closer agreement to the calculations of Wheeler and Lamb than to the estimate of Joseph and Rohrlich. In addition a short experiment was performed to measure the symmetry of the energy distribution between the electron and positron members of the pair. The results show that the energy-sharing curve is symmetrical as predicted by theory.

203. Mangum, Y. P., Jr.

FLUX FROM HOMOGENEOUS CYLINDERS CONTAINING UNIFORM SOURCE DISTRIBUTIONS. Newport News Shipbuilding and Drydock Co., Va. Rept. no. NNSD-R-7-4/1, 15 Nov 56, 41p.

Data are presented tabularly on flux from a cylindrical source attenuated by plane laminated shields. Plots of $\pi \nu \phi$ as a function of the shield thickness in units of the absorption mean free paths are presented where ϕ = flux (cm^{-2}-sec^{-1}), and ν = absorption coefficient of cylinder (cm^{-1}).

204. Metallwerk Plansee Gesellschaft m.b.H.

IMPROVEMENT IN OR RELATING TO MATERIALS FOR SHIELDING AGAINST RADIATION. BRITISH PATENT 861,657, 22 Feb 1961.

A simple method is given for manufacturing a uranium-tungsten alloy material for use as a shielding material in radioisotope containers. This method consists in compressing tungsten powder on or as the container and impregnating the compressed tungsten powder with molten uranium or a molten uranium alloy in vacuum or an inert atmosphere. The materials obtained have a density of $\sim$ 19 g/cm^3 and have good machinability. It is possible to produce materials with a uranium content up to $\sim$ 40% (using tungsten with 40% voids).

205. Mar, B. W.

THE BOEING SHIELDING RESEARCH PROBLEM. Boeing Airplane Co., Seattle. Rept. no. NP-10038, p. 424-31.

A description is given of the program which is designed to encourage fundamental research to supplement applied radiation shielding wo rk in nuclear weapons, reactors, and outer space.

206. March, J. P., et al

GAMMA RAY SCATTERING.

Seventh quarterly technical rept.

for period 18 June 60 through 18 Sep 60.

Florida Univ., Gainesville. Rept. no.

NP-9430, 12p. (Contract NOas 44013-c)

The angular distributions of radiations emerging from a lead shield and from polyethylene-lead shields were studied. A 1-curie Co^{60} source was used. The data are expressed as the ratio of the counting rate from one shield to the counting rate from the standard shield.

207. Marjon, P. L.

RADIATION PROTECTION CHARACTERISTICS OF PARTIAL SHIELDS FOR NUCLEAR AIRCRAFT SERVICING. Convair, Fort Worth, Tex. Rept. no. NARF-58-20T; MR-N-200; Project 6(1-9964), 21 Apr 58, 45p. (Contract AF33(600)-32054)

An experimental investigation was made to determine the limiting radiation field parameters in the use of the partial shield concept in ground support applications for nuclear aircraft. When used in the presence of multiple sources of radioactivity, such as an activated airplane, the partial shield can reduce the radiation exposure of a mechanic to that amount which originates in his immediate work area, by providing shielding on all sides except for an opening through which work is performed. Another feature of the partial shield concept is that it utilizes the higher total radiation dose which the hands are permitted to receive over the allowed total body tolerance. A reactor fuel element was used as a source to investigate the radiation protection characteristics of two elementary versions of the partial shield concept. One is the open-front type, which is simply a shield box with a work opening. The other is the body-shielded, open-front type. This type is similar to the open front version except that it has an additional body shield inside the box. Field conditions are defined in which direct maintenance would be possible on a nuclear airplane by use of these types of partial shields.

208. Mayburg, S. and Witzig, W. F.

THE GAMMA RAY SHIELDING OF IN-PILE EXPERIMENTS. Westinghouse Electric Corp. Atomic Power Div., Pittsburgh, Pa. Rept. no. WAPD-P-192, 12 May 52. Decl. 2 May 57, 15p. (Contract AT-11-1-GEN-14)

The possibilities of using a gamma ray shield to reduce gamma heating in an in-pile experiment has been examined for the Materials Testing Reactor. It is found that an optimum shielding material has a high density, high melting point, good thermal conductivity, and a small ratio of the thermal neutron cross section to atomic weight. Lead, while not ideal, fits these experiments better than most materials. Because an effective shield must be at least a centimeter thick and well cooled, the shield requires more space than is available within the active lattice of the MTR. However, sufficient space may be available in other experimental holes.

209. Mazari, M., et al

Coefficients of linear absorption of fast neutrons in concrete. (Instituto de Física de la Universidad Nacional Autonoma de México). REV. MEX. FÍS. v. 6, p. 1-8, 1957. (In Spanish)

Coefficients of linear absorption in samples of concrete, clay, and plaster were determined. In a series of 12 samples of concrete fabricated with normal and rapid setting cement, employing andesitic aggregates Σ was observed to diminish with density and increase with neutron energy. The Σ's vary from 0.085 to 0.097 cm for densities ρ ranging from 2080 to 2190 kgm^{-3}, using neutron energies of 13.0, 14.5, and 16.2 Mev. Determinations in a concrete slab 1m x 1m x 10.1 cm yield Σ values of the order of 58% with respect to a cylindrical tube of the same dosage. In a dry clay of Guaymas, and in common plaster, the linear absorption coefficients in the same energy range give results of 0.069 and 0.071 cm^{-1}, respectively. This information should be useful in the design of walls for protection against fast neutrons using concretes common in Mexico.

210. Meem, J. L. and Fairbanks, F. B.

SHIELDING REQUIREMENTS FOR THE ARMY PACKAGE POWER REACTOR.

Alco Products, Inc., Schenectady, N. Y.

Rept. APAE-3, 1 May 56. Decl. with deletions 7 May 57, 70p. (Contract AT-(11-1)-318)

The design, selection, and calculation of the Army Package Power Reactor shielding are described. The APPR-1, a prototype of a package reactor for remote locations, has a primary shield of Fe and water. This shield has been adopted to permit fast erection and to provide low transported weight. Economically, including transportation cost, the Fe-water shield is better than a Pb-water shield and is competitive with a concrete shield for a remote site. Because of its location at Fort Belvoir, Va., the shielding requirements for the APPR-1 are considerably more stringent than those for a reactor at a remote base. Since the secondary shielding which surrounds the entire primary system must provide protection for personnel at any location outside the vapor container, concrete is provided for this need.

211. Merrill, M. and Cowell, W. L.

LITERATURE SURVEY OF CONCRETES FOR NUCLEAR RADIATION SHIELDING.

Naval Civil Engineering Lab., Port Hueneme, Calif. Type C final rept., 28 June 60, 32p.

A literature survey was made to determine the extend of knowledge about shielding materials used as aggregates in concrete. Information is presented on mix designs incorporating various types of heavy aggregates, available cost data relating to construction methods and aggregates, problems associated with heavy concretes during construction, and nuclear shielding properties for concretes. The use of heavy aggregates for increasing the density of concretes has been well explored by previous investigations. At present there are no materials that can compete economically with barite and iron ores. Of the iron ores, magnetite is the most popular. Of the high cross-section materials for thermal neutron absorption, boron is generally the most economical.

212. Mezger, F. W.

Suppression of neutron-induced activation outside a hydrogen ous shield. General Electric Co., Cincinatti. NUCLEAR SCI. AND ENG. v. 4, p. 96-111, July 1958.

Two methods are evaluated for the suppression of neutron-induced activation outside hydrogenous shields: the addition of a l/v absorber to the hydrogenous shielding material, and the addition of a layer of l/v absorbing material to the outside of the shield. When the material being activated has a l/v cross section, a closed-form solution is obtained. The results should prove valuable in determining to what extent activation outside a shield can be reduced by any combination of these two methods.

213. Miller, J. M.

Effective neutron removal cross section of zirconium. Oak Ridge National Lab., Tenn. In Paper 5 of PAPERS FROM SEVENTH SEMI-ANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 4p.

The effective removal cross section of zirconium was determined as 2.36 ± 0.12 barns on the basis of thermal-neutron flux measurements made at the LTSF beyond two slabs of zirconium (1.8 wt.% hafnium), each 54 by 49 by 2 in. A mass attenuation coefficient (Σ_R/ρ) based upon the removal cross section and a measured density of 6.54 g/cm^3 is $(1.56 \pm 0.08) \times 10^{-2}$ cm^2/g.

214. Miller, R. A. and Cranford, W.

SHIELD SYSTEM OPTIMIZATION: A PARAMETRIC STUDY. Convair, Fort Worth, Texas. Rept. no. NARF-58-24T; MR-N-207; Project no. 6(1-9964), 16 June 58, 34p. (Contract AF-33(600)-32054)

214. (cont'd) In the first application of the gradient nonlinear programming procedure to optimization of the total weight of a divided shield system, a point source was assumed and crew compartment dose rate and reactor leakage constraints were applied. To make the results more realistic, the gradient nonlinear programming procedure was extended so that the effects on shield shape of different locations of a point source and a point detector can be studied. The weight function is minimized subject to a nonlinear dose rate constraint, reactor leakage constraints, and maximum radius constraints--all of which are expressed as linear inequalities. Included here is an outline for the mathematical formulation of the procedure to be used in the numerical computations.

215. Mironov, S. (C.) A., and Ponomarev, K. K.

Concrete for protection against ionizing radiation. (Béton de protection contre les rayonnements ionisants). CEA-tr-R-856. (Translated into French from BETON I ZHELEZOBETON p. 259-62, 1956.

The principles underlying the selection of materials for biological shields are reviewed, and different types of shielding concretes are described: barite concretes, magnetite concretes, concretes with limonite, ordinary concrete, and concretes with special cements.

216. Solar flares pose weight problem.

MISSILES AND ROCKETS v. 8, n. 2, p. 18, 9 Jan 1961.

A plan is proposed to use ablative materials, propellants, equipment and structure, as shielding from hazardous solar flares, to reduce the weight in space vehicles.

217. Moncassoli, A. M. and Massera, A.

CRITERIA FOR SELECTION OF PRIMARY AND SECONDARY SHIELDING FOR A NAVAL REACTOR. "VI Rassegna Internazionale Elettronica e Nucleare. Atii

217. (cont'd) Del Congresso Scientifico: Giugno 1959. Sezione Nucleare. Volume 1." Rome, Comitato Nazionale Ricerche Nucleari, p. 273-98, 1959. (In Italian)

Criteria for selecting shielding for a naval PWR are pointed out, and the calculations involved in making the choice are shown. The results are tabulated.

218. Montague, R. M., Jr.
ARMY RADIOLOGICAL SHIELDING REQUIREMENTS. Army, Washington, D. C. Rept. no. NP-10038, p. 9-13.

A discussion is given of what the Army will look like in 1965 to 1970; the doctrine, equipments, and tactics. Onto this framework, estimates are given of the general shielding requirements.

219. Morabito, J. J.
SHIELDING ON 30-INCH PIPE. Atomic Power Development Associates, Inc., Detroit. Technical memorandum no. 10; rept. no. AECU-4095, 5 Feb 58, 9p. (Contract AT(11-1)-476)

The materials of construction and the design of the shielding for a 30-inch pipe are presented. A calcium borate and a borated diatomaceous earth-lumite concrete performed equally satisfactorily and both are considered for the shielding of the pipe.

220. Morgan, R. S.
LEAD CONTAINERS FOR RADIOACTIVE ISOTOPES: THEIR WALL THICKNESS AND

220. (cont'd) ITS CALCULATION. Gr. Britain. Armament Research and Development Establishment, Fort Halstead, Kent, England. Rept. no. ARDE(M)-24/57, May 57, 16p. ASTIA AD-136 006

The design of source containers with adequate safety without sacrificing portability is discussed.

221. Morgan Crucible Co., Ltd. New material for the simultaneous slowing down and absorption of neutrons. GERMAN PATENT DAS 1,043,534, 13 Nov 58. ATOMPRAXIS v. 5, p. 501, 1959. (In German)

A shielding material for neutrons is discussed. It is composed of carbon and a boron compound and possesses at high refractoriness and heat conductivity a maximum moderating effect per unit space. Fast neutrons should be completely slowed down by the material and converted into heat energy and slow neutrons. Simultaneously the neutrons should be distributed homogeneously through the material and be absorbed. The material is formed from a mass of carbon, preferably graphite, in which a boron compound is distributed, which is present in the form in which it is solidified from the molten state in situ in this mass. The boron compound can be contained therefore in numerous pores in carbon matrix. A detailed method for the production of this material is given.

222. Morton, K. W. Scaling neutron tracks in Monte Carlo shielding calculations. Atomic Energy Research Establishment, Harwell, Didcot, Berks. J. NUCLEAR ENERGY v. 5, p. 320-4, 1957.

222. (cont'd) A simple technique is given to enable neutron tracks generated in a Monte Carlo study of a given slab of material to be used for slabs of any other thick ness. It is applied to the interpretation of some thick sample scattering data.

223. Moulthrop, H. A.

Z PLANT RADIATION STUDY INTERIM REPORT NO. 5. PART II. DATA ON GAMMA SHIELDING OF SPECIAL PLUTONIUM SAMPLES. General Electric Co., Hanford Atomic Products Operation, Richland, Wash. Rept. no. HW-61755, 22 Oct 59, 103p. (Contract W-31-109-eng-52)

The effectiveness of lead, lead glass, steel, safety glass, and plexiglas for shielding gamma radiation from plutonium samples, including powders, solution, and metal, was evaluated.

224. Muckenthaler, F. J. and Blosser, T. V.

DESIGN OF A SHIELD FOR THE ORR SOUTH FACILITY: EXPERIMENTAL PROGRAM AT BEAM HOLE HB-2. Oak Ridge National Lab., Tenn. Rept. no. ORNL-3016, p. 221-6.

The thermal-neutron flux, fast-neutron and γ doses, and fast-neutron and γ ray spectra from beam hole HB-2 of the ORR were measured. The data will be used in the design of a portable shield for the ORR South Facility in connection with the installation there of the GCR loop no. 2. Experimental configurations included barytes concrete, iron, lead, polyethylene, and borated polyethylene, used either singly or in various combinations. An attempt was made to evaluate the dose from scattering by these materials. Only a sampling of the data is presented.

225. Murray, G. S., Roberts, R. and Dove, D.

THE DESIGN AND CONSTRUCTION OF LARGE GAMMA IRRADIATION FACILITIES. (Presented at the International Conference on Radioisotopes in Scientific Research, 9-20 September 1957 at Paris). Rept. no. UNESCO/NS/RIC/19. London, Pergamon Press, Ltd., 1957, 11p.

The design of radiation shielding suitable for sources of many thousands of curies of gamma activity is discussed. The main object in the design has been found advantageous to be able to carry out irradiations over a wide range of physical conditions, such as temperature and pressure, and also to impose the minimum restriction on the size of vessel being irradiated. The requirements have been met in an entirely new design of irradiation unit. The advantages of a large concrete shield compared with the usual lead shield are discussed. A particular feature of the design is the achievement of a variable source geometry, so that a source of any shape can be built up within a large irradiation chamber. Attention is drawn to the need for simplicity in operation while achieving maximum protection for personnel using the source. Details are given of interlock mechanisms, and also the dose rates obtained at various positions in completed shield. Some figures for the cost of construction of large radiation sources are also included.

226. Murphy, G.

NEUTRON ATTENUATION IN LAYERED IRON-WATER SYSTEMS. Iowa State Univ., Iowa City. Rept. no. NP-10038, p. 468-73.

Discussions are given of the applications of Monte Carlo techniques to the determination of neutron distributions in a series of alternating semi-infinite slabs of iron and water, and experimental work in the study of neutrons in iron-water systems.

227. Musialowicz, T. and Wachsmann, F.

Shielding beta radiation. (Universität, Erlangen, Germany). ATOMPRAXIS v. 6, p. 404-7, Oct/Nov 1960. (In German)

One of the questions involved in shielding β rays is the amount of x radiation produced in the cover film. The ratio between the β dosage rate and the x-ray dosage rate is especially important in shielding the eyes. Since the proportion of x rays produced in the absorber increases with its atomic number, there was doubt as to the usefulness of using lenses with a higher efficient atomic number in safety goggles for protection against β radiation. Calculations and empirical measurements both showed that the atomic number of the shielding material makes no difference, since even with lead the proportion of x rays is less than 0.1%.

228. Muta, N. and Tadashi, S.

A test on paint for protection against stray radiation. Sapporo Medical College. SAPPORO IGAKU ZASSHI v. 17, p. 148-9, Feb 1960. (In Japanese)

A paint for protection against stray radiation was tested. The paint in question consists of three coatings, namely, lead suboxide as a basic coating, iron oxide as an intermediate layer, and titan white as the top coating. It was found that, when the base board was merely plywood, the amount of stray radiation is higher with the coating than without. When 12-mm boards, 25-mm boards, and a 3-mm lead sheet were used as the bases, the protection values were 2.0, 2.8 to 2.9, and 1.3, respectively; these values, however, were not so high as those reported by Hashizume.

229. McCall, R. C.

Low-background shields. General Electric Co., Richland, Wash. HEALTH PHYS. v. 2, p. 304-6, Feb 1960.

A shielded cell around a Na(T) crystal scintillation detector is described which was constructed of polyethylene-lined paper bags containing 100 pounds each of crushed talc. The cell was 4 ft x 4 ft x 4 ft with 2-ft thick walls. Background in the talc

229. (cont'd) cell ranged from about 200% at low energies to 160% at high energies of that obtained in a lead cell. The cost of the talc cell was about 20% of that of a lead cell. Use of the cell for counting whole-body gamma radiations in humans is described.

230. McCool, W. J. and Otis, D. R.

Calibration of ORNL Lid-Tank fission plates. Pratt and Whitney Aircraft Div. of United Aircraft Corp., Middletown, Conn. and Convair, San Diego, Calif. NUCLEONICS v. 18, n. 4, p. 98, 100, 122-3, Apr 1960.

The SP-2 fission plate in the ORNL Lid Tank Shielding Facility was calibrated on three occasions and these data were used for a retrospective recalibration of SP-1, whose nonuniform geometry made its original calibration suspect. The three independent calibration experiments performed to determine the fission rate in SP-2 were: thermal-neutron flux and utilization; neutron-generation rate; and heat-generation rate. A weighted average of these methods gives a fission rate of 1.62 x 10'' ± 5% fissions/sec of equivalent to 5.18 ± 5% watts of reactor power. The weighted average effective source strengths of LTSF fission plates are 1.7 and 1.14 watts for neutrons and gamma in SP-1 and 4.87 and 4.56 watts in SP-2. The large difference in the effective power of SP-1 is attributed to excessive self-absorption of gammas by the thick SP-1 components.

231. Nelkin, M.

The decay of a thermalized neutron pulse. General Atomics Div., General Dynamics Corp, San Diego, Calif. NUCLEAR SCI. AND ENG. v. 7, p. 210-16, Mar 1960.

A theoretical basis is presented for the interpretation of the pulsed-neutron technique for measuring thermal-neutron absorption cross sections and transport parameters. A procedure is given for the exact solution of the Fourier-transformed, multivelocity transport equation in an infinite medium. The method used is an expansion of the decay constant and neutron spectrum in a power series in the Fourier-transform variable. The procedure is first illustrated for the case of isotropic scattering and then generalized to anisotropic scattering by using the sperical harmonics expansion.

231. (cont'd) The results are given in terms of integral equations whose solution involves a knowledge of the energy-transfer cross sections between thermal neutrons and the moderating material. The approach employed is to extract the maximum amount of information which is dependent of these cross sections and to derive explicitly the equations involving them. The relation of the infinite medium Fourier transform variable to the geometric buckling of a finite sample is discussed. The validity of the interpretation of the experiments in terms of the diffusion coefficient by reducing the assignment of an equivalent infinite medium buckling to each finite sample measured is discussed.

232. Nichols, L.D.

EFFECT OF SHIELD POSITION AND ABSORPTIVITY ON TEMPERATURE DISTRIBUTION OF A BODY SHIELDED FROM SOLAR RADIATION IN SPACE. APPENDIX A - DERIVATION OF ANGLE FACTORS. APPENDIX B - LIMITING CASES. APPENDIX C - MAXIMIZATION OF SHIELD TEMPERATURE WITH RESPECT TO ABSORPTIVITY. Rept. no. U.S., NASA TN D-578, Jan 61, 41p.

Analysis of temperature distributions on two disks subjected to solar radiation. With one disk shielding the other, the effects of the temperature distribution of absorptivity, thermal conductivity, and spacing between the disks were studied. The calculations show the possibility of using a movable shield as a temperature control device for a space vehicle.

233. Nicholas, M.S., Jr.

THE ABSORPTION AND SCATTERING OF RADIATION IN RANDOM AGGREGATES OF PEBBLES. Oak Ridge National Lab., Tenn. Rept. no.

233. (cont'd) CNL-21, Decl. 1 Sep 60, 64p. (Contract W-7405-eng-26)

The attenuation of a collimated monoenergetic beam of radiation in traversing a random aggregate of pebbles is considered. It is found that there is a transmission of radiation over and above that allowed by the main density of material and that transmission is essentially exponential. This implies that a shield composed of pebbles must be increased in depth by a given ratio over that computed by means of the average density alone. A formula is given for the approximation of this ratio. The problem of computing the channelling effect reduces itself to the problem of determining the probability of penetrating an accumulated depth x' of solid material in traversing a total distance x in the pebble aggregate. The theory for this computation is developed in general for cases of continuous distribution of solid and void segments and for cases where these distributions have only discrete values. The theory for the discrete distributions is developed with the size of the discrete subunit of length as a parameter. The effect applies not only to attenuation but also to diffusion; that is, it is a purely geometrical effect. This follows from the fact that the probability of scattering or absorption is found by weighting the probable amount of solid penetrated by a negative exponential; that is, by the appropriate transport-diffusion kernel. Therefore the discussion applies not only to shielding but to all problems dealing with the transmission and diffusion of radiation through a bed of randomly oriented pebbles and is of sufficient magnitude to warrant attention whenever the pebble size is comparable to the relaxation length. Whenever the interstices between the pebbles are filled, the channelling effect still exists, but in a reduced amount. A method of estimating the effect for this case is given.

234. Niemann, E.G.

STUDY OF PROTECTION AGAINST RADIATION IN THE FIELD OF CONSTRUCTION. (Etudes de la protection contre les rayonnements dans le domaine des matériaux de-construction). (Presented at Conference on Atomic Piles for Naval Construction, held in Hamburg, 25-28 November 1959.) Rept. no. CEA-tr-A-724. (Translated into French by P. Monard)

Several studies made on different types of concrete are discussed with respect to their use for protection against radiation. The effects of geometry, energy spectrum, and neutron activation are considered.

235. Noyes, J.C. and Brown, W.D.

SHIELDING REQUIREMENTS FOR

RADIATION BELT PARTICLES.

Boeing Airplane Company, Scientific

Research Laboratories. Rept. no.

BOAC-DI-82-0048; D1-82-0048, Jan 60.

The proton energy spectrum in the heart of the inner radiation belt is presented. The degraded spectra resulting from the attenuation of the incident protons by 5, 10, and 20 gm/cm^2 of polyethylene are calculated. Air dose rates ranged from 54 r/hr for no shielding to 2 r/hr for 20 gm/cm^2 shielding.

236. Radiation protection. New approach

to measurements needed. NUCLEAR

ENG. v. 5, p. 212-13, May 1960.

Effects of the incidence angle on the attenuation of gamma radiation by lead, concrete, and polyethylene barriers were studied using Co^{60}, Cs^{137}, and Au^{198} sources. The data are expressed in terms of ratios of the observed dose rate to that calculated on the basis of normal incidence data for a barrier thickness equal to the oblique path length. The ratios were nearly 1.0 at small oblique angles, but increased with the angle. This effect is greatest for low-energy radiation and thick barriers.

237. Nuclear Shielding Supplies and Service,

Inc., White Plains, New York

RADIATION SHIELDING --BIBLIOGRAPHY.

1959, 27p.

A bibliography of 331 references on shielding is presented.

238. Oberhofer, M. and Springer, T.

Shielding of gamma radiation. Radiation

protection in reactors and in industrial

238. (cont'd) nuclear plants. (Technische Hochschule, Munich). KERNTECHNIK v. 2, p. 124-6, Apr 1960. (In German)

The fundamentals for the rough calculation of gamma shielding are presented. The absorption law, the half-value layer and 1/10 value layer, and the shielding of gamma radiation by thick absorbers greatly expanding laterally are discussed.

239. Office of Civil and Defense Mobilization, Battle Creek, Michigan and Naval Radiological Defense Lab., San Francisco. NRDL--OCDM SHIELDING SYMPOSIUM PROCEEDINGS, 31 OCTOBER - 1 NOVEMBER 1960. Reviews and lectures no. 110, 501p.

Presented are the proceedings of the NRDL-OCDM Shielding Symposium, which was organized to provide a review of work going on, the state of the shielding art, and future needs toward development of a broadly applicable shielding technology. Separate abstracts were prepared for 29 papers.

240. Osanov, D.P. and Kovalev, E.E. The radiation from a cylindrical source behind a flat shield. ATOMNAYA ENERG. v. 8, p. 374-6, Apr 1960. (In Russian)

Radiation produced in multiple scattering (using aqueous cobalt sulfate solutions) in front of a shield of $\mu_2 d > 2$ to 2.5 was a small fraction of that behind the shielding. The dosage from multiple scattering may be calculated using the equivalent absorption length and the point source accumulation factor. The maximum difference between experimental and calculated data with $\mu_2 d < 2$ to 2.5 did not exceed 30%.

241. Pagnelli, A. PROTECTION SCREENS AGAINST THERMAL NEUTRONS AND ITS

241. (cont'd) MANUFACTURING PROCESS. BELGIUM PATENT 580,330, 4 Aug 1959. (In French)

Instead of manufacturing expensive BC plates, granules of B or BC are incorporated in a suitable metallic or nonmetallic binding material, heated, and sprayed onto the surface to be shielded from either γ rays or neutrons.

242. Parle, W.C. and Erskine, A.M. Paraffin wax may be better for shielding. Researchers feel hydrogen atoms in wax could absorb more neutrons than plastics. California Ink Co., Berkeley. MISSILES AND ROCKETS v. 5, n. 11, p. 34, 16 Mar 1959.

The development of lead-dispersed-in-wax shielding and its manufacture are described. The physical properties and shielding effectiveness of this material are given.

243. Patoski, V.A. and Marjon, P.L. PARTIAL SHIELD REMOTE MAINTENANCE EXPERIMENT. Convair, Fort Worth, Texas. Rept. no. NARF-58-23T; MR-N-214; Project no. 6, 9 July 58, 30p. (Contract AF33(600)-32054)

An exploratory investigation was made to determine the feasibility of applying the partial shield concept to airplane servicing. The objective of this experiment was to determine the extent of the development effort which would be needed to implement the partial shield concept and to indicate major problem areas. In this experiment, a mechanic using long-handled tools performed several inspection and maintenance tasks on a B-47 airplane from within a partial shield mock-up. It was found that a mobile partial shield can be used at some locations and for some tasks on a contemporary airplane. Although this experiment was quite limited in scope, it showed that partial shields are feasible. Furthermore, experience was gained which will aid in establishing the development requirements for an engineered system.

244. Pearse, R.A.

ANALYSIS OF THE COOLING OF THE THERMAL REACTOR SHIELD. Great Britain. Department of Atomic Energy, Industrial Group H Q., Risley, Lancs, England. Rept. no. CF-5213; TRDC-P-32, 22 Apr 54, 36p.

The shield of the Thermal Reactor is heated by nuclear radiation and by heat transfer from the pressure vessel. Atmospheric air can be effectively used as a coolant. This note examines such an air cooling system in which the total air requirement is 84,400 C.F.M.

245. Penny, S.K.

PROPOSED CALCULATIONAL PROCEDURE FOR THE ORR REACTOR BEAM-HOLE SHIELDING. Oak Ridge National Lab., Tenn. Rept. no. CF-58-1-102, 28 Jan 58, 6p. (Contract W-7405-eng-26)

The equations, numerical values, and results of the calculation are given. The assumptions involved and the known uncertainties are listed.

246. Peterson, E.G.

SHIELDING PROPERTIES OF ORDINARY CONCRETE AS A FUNCTION OF TEMPERATURE. General Electric Co. Hanford Atomic Products Operation, Richland, Wash. Rept. no. HW-65572, 2 Aug 60, 42p. (Contract AT(45-1)-1350)

246. (cont'd) The effect of high temperature on neutron attenuation in concrete was determined. The fast neutron attenuation was exponential and could be adequately predicted using removal theory. The calculated removal cross sections compared with the measured values to within 3%. Tests indicated that even after penetrating 47.5 in. of concrete the thermal and resonance neutron fluxes had not reached equilibrium with the fast neutron flux. Significant changes in neutron flux values were observed after the concrete was baked at 100, 200, and 300°C. A combination of removal theories adequately predicted the increase in the fast, resonance, and thermal neutron fluxes. The measured relaxation lengths and removal cross sections are summarized. Changes in reflector and thermal shield thickness or composition from test values would affect the relaxation length. The shield facility was calibrated so that removal cross sections can be measured. The dose rate generated through the full thickness of the as-cured specimens was primarily due to the fast neutrons. After the concrete was baked at 300°C, the dose rate was a result of resonance neutron leakage. The γ dose rate was measured through 47.5 in. in the 100°C test only. The data indicated that the biological dose rate through 47.5 in. of concrete is primarily due to γ leakage at this temperature.

247. Peterson, E.G.

SHIELDING PROPERTIES OF FERRO-PHOSPHORUS CONCRETE AS A FUNCTION OF TEMPERATURE. General Electric Co. Hanford Atomic Products Operation, Richland, Wash. Rept. no. HW-64774, 15 July 60, 82p. (Contract AT(45-1)-1350)

An investigation was conducted to determine the shielding properties of ferrophosphorus concrete at temperatures up to 320°C for neutrons and gamma radiations. Ferrophosphorus concrete was selected because of its high density and possible use in biological-shield construction. The neutron and gamma attenuation properties of ferrophosphorus concrete were determined experimentally. The thermal and resonance fluxes are based on gold-foil activation while the fast flux was determined using the S^{32} (n, p) P^{32} threshold reaction. The gamma attenuation was measured with ionization chambers. The tests indicate that the ferrophosphorus concrete could be used as a biological shield. It should be pointed out, that under the conditions under which the ferrophosphorus concrete was tested, there are other concretes that could be used which would be more efficient as well as more economical.

248. Pierce, J.A., Jr., Captain; and James, P.M.

TOTAL LINEAR ABSORPTION COEFFICIENTS

248. (cont'd) OF VARIOUS TYPES OF MEDIA FOR GAMMA RADIATION. Army Chemical Center, Radiological Division, Maryland. Interim rept. TCIR 591, 8 Nov 51. ASTIA ATI 157 797

Absorption coefficients of various materials for which no data is available in literature are given.

249. Pirie, A. and Drance, S.M. Modification of x-ray damage to the lens by partial shielding. Univ. of Oxford. INTERN. J. RADIATION BIOL. v. 1, p. 293-304, July 1959.

We have examined the effect of irradiation of half the lens of the eye of the rabbit on the development of cataract and on the histology of the lens epithelium. Shielding one-half of the lens lessens the damage caused by irradiation in the other half. This lessening of damage, which is made evident by a less severe cataract, fewer nuclear fragments, smaller mitotic overshoot and reduced subsequent cell loss in the irradiated epithelium, does not appear to be due to cell migration from healthy to irradiated area. We suggest that lessening of the damage is caused by diffusion of substances into the irradiated area from other parts of the lens or eye.

250. Plessey Company, Ltd. CERAMIC MATERIALS FOR RADIATION SHIELDING. BRITISH PATENT 811,782, 15 Apr 1959.

A ceramic shielding tile containing Pb and Ba is described. In one composition, 25 parts $BaS0_4$, 75 parts $PbSi0_3$ and 1/2% bentonite are pasted, molded, and fired at 800°C. If the bentomite is omitted and it is fired at 1100°C a material suitable for radiation of 75 kev or less is obtained. Another composition is given which was compared with lead for broad and narrow beam x rays: $BaOCO_2$, 27 parts; Si02, 8 parts; Pb0 $(Si0_2)_2$, 48 parts; and $Zr0_2$, 17 parts. This composition was fired at 1110°C. A 1-cm thickness of this ceramic was equivalent to ~ 1.6 mm Pb at zero distance, increasing to 1.9 mm

250. (cont'd) Pb at 50 cm. These ceramic shielding tiles may be lead glazed to present an easily cleaned presentable surface, and they may be bonded together by a litharge cement.

251. Polano, G.

Criteria inherent to the basic calculation of primary shieldings for nuclear reactors. ING. NUCLEARE v. 3, p. 239-46, Nov/Dec 1960. (In Italian)

In order to limit as far as possible the work required in the calculation of shielding initial projects must rest on semi-empirical methods, offering sufficiently precise data to enable decisions as to types of shielding to employ. Considering the fact that the primary shielding must reduce neutron speed and gamma radiations, methods are given for calculating attenuation of the fast neutron flux; thermal neutron flux needed for the evaluation of the captured gammas; attenuation of the gammas.

252. Preiser, S. Mittelman, P.S. and Berndtson, C.R.

PLANE ISOTROPIC GAMMA RAY BUILDUP FACTORS IN LEAD AND WATER WITH APPLICATIONS TO SHIELDING CALCULATIONS. Nuclear Development Associates, Inc., White Plains, N.Y. (For Pratt and Whitney Aircraft Div.) Rept. no. NDA-10-144, 21 Dec 54, 17p.

Plane isotropic dose (B_r) and energy absorption (B_a) buildup factors were calculated for gamma rays in lead and water for various initial energies from 0.5 to 10 Mev and for penetrations up to 15 mean free paths. These buildup factors, computed by numerical integration of point isotropic buildup factors, are compared with plane isotropic buildup factors computed using two approximate methods. Specific examples of the use of plane isotropic buildup factors in simple shielding geometries are given.

253. Price, B. T., Horton, C. C. and Spinney, K. T.

Radiation shielding. (International Series of Monographs on Nuclear Energy: Division 10) REACTOR DESIGN PHYSICS v. 2, 350p. London; New York, Pergamon Press, 1957.

Stressing the physical principles underlying the engineering design of a reactor shield, this textbook treatment of radiation shielding empasses reactors, betatrons, syncrotrons, and linear accelerators; however, the shielding of very high-energy accelerating machines has been omitted. The various topics under discussion include the biological hazards of nuclear radiations, the attenuation of gamma rays and high-energy electrons, shielding against neutrons and neutron physics, neutron attenuation in thick shields, and the shielding of reactors and radioactive materials. The text is supported by numerous charts, diagrams, tables and bibliographies. The authors are all with the Atomic Energy Research Establishment at Harwell.

254. Pullman, L.

THE TRANSMISSION OF GAMMA RAYS AT SLANT INCIDENCE THROUGH SLABS OF VARIOUS MATERIALS IN SHIELDING. Nuclear Development Corp. of America, White Plains, N.Y. Rept. no. NDA-2056-6, 10 Oct 56. Decl. 31 Mar 60, 109p. (For General Electric Co. Aircraft Nuclear Propulsion Department. Subcontract AT-29)

Compilations of and analytical indexes to graphs and tables of γ ray slant penetration data and graphical comparisons between attenuations computed from Reactor Handbook γ ray slant penetration parameters, and selected attenuation data reported elsewhere are presented.

255. Raleigh, H. D.

RADIATION SHIELDS AND SHIELDING.

A LITERATURE SEARCH. AEC,

Technical Information Service Extension.

Feb 60, 56p.

A literature search, containing 659 references, was prepared to serve as a reference guide until the completion of a formal bibliography supplement. References pertain chiefly to reactor shielding, but references on gamma and neutron attenuation, properties and effectiveness of shielding materials, biological shielding, shielding design and theory, etc., are also included.

256. Randall, C.H.

THE CENTRAL PROBLEM OF

DISPERSION ANALYSIS. Knolls

Atomic Power Lab., Schenectady, N.Y.

Rept. no. KAPL-M-CHR-4, 20 Feb 61,

24p. (Contract W-31-109-eng-52)

The performance of a dispersion material generally depends on its microstructure. In order to evaluate such sensitivity it is necessary to construct a mathematical model to describe the microstructure. A simple example of such a model is obtained by distributing particles of one phase randomly in cubic cells of another. As a consequence of this model, phase intercept distributions and neutron self shielding factors may be estimated. Another consequency of this model is a coarse but simple test for particle clumping. It can be shown that the probability of finding a w_0 by t_0 rectangular field, devoid of particles, on K, W by T, photomicrographs is P_0; if $P_0 << 1$ and

$$(w_0\, t_0) \cong \ln\left[\frac{-\mathrm{Ln}\,(1 - P_0)}{\frac{V}{\beta}\,\frac{WT\beta}{\bar{a}} + 1\,K}\right]\left[\left(\frac{\beta}{\bar{a}}\right)\ \mathrm{Ln}\,(1 - \frac{V}{\beta})\right]^{-1}$$

Here V is the particle volume fraction, β is the areal packing factor and $\bar{a}$ the particle mean areal intercept.

257. Rasch, R.

RADIATION SHIELDS. BRITISH

PATENT 870,106, 14 June 1961.

A radiation shield is described. The shield is composed of 80 to 95% by weight ferro-silicon, in the so-called granular iron form, in cement. The particle size of the ferro-silicon material is broken down for the make-up of the shielding material.

258. Raso, D.J.

Transmission of scattered γ rays

through concrete and iron slabs.

Technical Operations, Inc., Burlington,

Mass. HEALTH PHYS. v. 5, p. 126-41,

June 1961.

The penetration of γ rays undergoing one or more scatterings in concrete and iron of various slab thicknesses (3.75 to 60.0 g/cm^2) was investigated by the Monte Carlo technique. Radiation with energy of 350, 660, and 1250 kev and also the γ spectra produced by the products (omitting volatile materials) of U^{235} fission at 1.12 and 23.8 hr after fission were assumed to be incident on the medium at angles of cos $\theta_0 = 0$, 0.25, 0.50, 0.75, and 1.0. The case histories of 5000 photons were followed on the IBM 704 digital computer. The information obtained included: number and energy transmitted, the energy distribution of the transmitted radiation integrated over all directions, the angular distribution integrated overall energies, and the angular distribution of the transmitted energy and transmitted "dose." These computed quantities refer to transmitted current, and an approximation is given for determining flux density. Comparisons are made with other theories and an experiment performed at USNRDL.

258A. Redmond, R. F.

SPACE RADIATION AND ITS EFFECTS

ON MATERIALS. Radiation Effects Infor-

mation Center, Columbus, Ohio. REIC

memo no. 21 (supercedes REIC memo no. 19),

30 June 61, 34p. (Contract AF 33(616)7375,

258A. (cont'd) Proj. 1448) ASTIA AD-261 277

The status of space radiation research in relation to its effects on materials and man is discussed from the point of view of recent literature. Theories concerning the Van Allen belts, cosmic radiation (galatic and solar cosmic rays), interactions of space radiations with matter (electrons, bremsstrahlung, and protons), and radiation, in particular as it effects man and semiconducting devices, are reported.

259. Reinig, K.

IMPROVEMENTS IN OR RELATING TO RADIATION-ABSORBING PROTECTIVE MATERIALS AND PROCESSES FOR THE FOR THE PRODUCTION OF SAME.

BRITISH PATENT 851, 602, 19 Oct 1960.

A resilient radiation-absorbing protective material for x-rays and ultra short wave radiation may be fabricated by using a fat or synthetic rubber incorporating a lead compound with an acid radical and an optical brightening agent. The lead compound may be lead nitrate or lead sulfate, and the brightening agent may be one used for textiles, e.g., Blankofor WL. The material may be used as a paint to coat walls (e.g., of packages or vehicles for transporting radioactive materials) or to impregnate fibrous carriers (e.g., gloves). In the latter case, if the carrier is inflammable, it is advantageous to incorporate a flame-proofing agent like Akaustan A. The material may also be applied as a skin cream to the body parts to be protected from radiation. The procedure for preparing the material is given.

260. Reissner, E.W.

IMPROVEMENTS IN OR RELATING TO SCREENS ADAPTED TO GIVE PROTECTION AGAINST NEUTRONS. Factories Direction, Ltd. BRITISH PATENT 865,435, 19 Apr 1961.

Screens which give protection against thermal and fast neutrons may be made comprising layers of hydrogen carrier interleaved with boron-containing layers. In one preferred embodiment of the fabrication procedure, the hydrogen carrier is a paper film impregnated with synthetic phenol-formaldehyde resin, and the boron-containing layers are layers of boric oxide bound in thin resin layers. Details of the fabrication processes are given.

261. Reitz, D.

Cosmic rays, nuclear reactors, and manned space systems. Martin Co., Denver. AERO-SPACE ENG. v. 20, n. 4, p. 28-9, 77-94, 96, Apr 1961.

Time variations and steady-state cosmic radiation conditions are presented. Steady-state biological intensities are given both for low-altitude orbits as functions of orbital inclination to the plane of the earth's equator, and for paths lying far above the earth's surface. Total allowable nuclear reactor radiations in the cosmic-ray background are next calculated as functions of satellite orbital inclination and exposure time. These reactor radiation intensities give no more than the recommended maximum doses as stipulated by the U. S. Committee on Radiation Protection. A method of obtaining minimum weight reactor shields also is outlined and applied in a hand computation for a nuclear 300-kw electrical system.

262. Robey, D. H.

ON THE INFLUENCE OF PARTICLE RADIATION ON MANNED SPACE FLIGHT. Preprint 230R. SAE-AFOSR, Astronautical Symposium, Los Angeles, Calif., 12-14 Oct 1960.

Discussion of the various types of radiation in space, together with calculations of the shielding necessary for protection to crews of space vehicles. The largest observed solar flare produced radiation against which a carbon shield weighing 140,000 lbs. probably would be required. Recommendations as to orbit and altitude for the minimization of radiation hazards are also made. The possibility of a system giving advance warning of unusually intense radiation is briefly discussed.

262A. Robey, D. H.

Radiation shelters for lunar exploration. J. ASTRON. SCI. v. 8, p. 62-63, Summer 1961.

262A. (cont'd) Discussion of the radiation hazards to human life, from solar flare and cosmic ray particles, occasioned by the absence of a natural radiation shield on the moon. It is suggested that human explorers prepare emergency flare shelters of lunar material.. A thickness of 50 to 100 cm. is found to be sufficient for shielding from most flares, assuming that the lunar surface material has a specific gravity of about two.

263. Robey, D.H.

Radiation shield requirements for two large solar flares. Convair, San Diego, Calif. ASTRONAUT. ACTA v. 6, p. 206-24, 1960.

The class 3+ flare of May 10, 1959, and the class 4 flare of February 23, 1956, were analyzed to determine the magnitude of radiation shielding required for man in space. The former was probably the largest flare of the year and the latter the largest flare on record. Carbon was selected for the shielding material. A spherical carbon shield configuration with an inside cavity of 90 cm radius was used for both events. It was found, that for the class 4 flare, all protons with energies below 1.4 Bev must be stopped in order to reduce the proton radiation dose to the order of 25 rem (RBE = 1). An optimistic estimate of the shield weight gave about 379 metric tons (834,000 Lb), neglecting nuclear reactions.

264. Robson, J.M.

THE ATTENUATION OF NEUTRONS BY THE SIDE SHIELD OF THE NRU REACTOR. Atomic Energy of Canada Ltd., Chalk River, Ont. Rept. no. CRP-860; AECL-932, Oct 59, 25p.

Results of measurements of the thermal, epithermal, and fast neutron fluxes at various locations in the side shield of the NRU reactor are presented. Mean free paths for those neutrons whose energy exceeds 2 Mev are found to be 8.3, 5.5, and 13.4 cm in light water, steel, and ilmenite concrete, respectively.

265. Roembke, J. E.
PROBLEMS ENCOUNTERED IN THE TEACHING OF SHIELDING ANALYSIS AND DESIGN TECHNIQUES. Office of Civil and Defense Mobilization, Battle Creek, Michigan.

266. Sanders, A. P.
Roentgen or gamma ray shielding in radiation protection. Duke Univ., Durham, N. C. AM. J. ROENTGENOL. RADIUM THERAPY NUCLEAR MED. v. 79, p. 532-6, Mar 1958.

267. Saxe, H. C.
THE PHYSICAL PROPERTIES OF BARYTES-COLEMANITE CONCRETE AND BARYTES-COLEMANITE-WATER MIXTURES. Georgia Inst. of Tech., Atlanta. School of Civil Engineering. Rept. AECU-3617, Feb 55, 59p. (For Oak Ridge National Lab.) (Contract W-7405-eng-26, Subcontract no. 715.)

This report is divided into two distinct parts, namely: Part I, Physical Properties of Barytes-Colemanite Concrete, and Part II, Physical Properties of Barytes-Colemanite-Water Mixtures. Part I contains a detailed study of the effects of colemanite admixture upon the setting and strength characteristics of barytes concrete for

267. (cont'd) nuclear shielding purposes. Two grades of colemanite ore were used in the study, each mainly differing in grain size distribution. The initial study, performed on the finer graded colemanite, indicated that serious deleterious effects upon the setting and strength characteristics of the concrete were experienced, unless some type of accelerator was used in conjunction with the mix. Later studies, however, indicated an absence of these deleterious effects if a coarser graded colemanite was utilized. The study includes recommendations as to batch proportions, and types of accelerators to utilize, and also contains data as to the strength and densities obtainable using these mixtures. Part II is devoted to the study of the properties of compacted barytes-colemanite-water mixtures as a substitute for barytes-concrete shielding. The work includes recommendations regarding the optimum unit weights obtainable, the desired batch proportions for optimum conditions, and the mixing and compaction procedure to be followed. It also includes estimates of the angle of internal friction φ, and the angle of wall friction upon carbon steel plate φ, to be used in calculation of lateral thrusts exerted by the confined material using Janssen's Formula.

268. Schroeder, E.

Calculations of temperatures and thermal strains in shielding. (<u>Calcul des temperatures et des contraintes thermiques dans les protections</u>). CEA-tr-A-903. Translated into French by Z. Tilliette from ATOMKERN-ENERGIE v. 4, p. 442-5, 1959.

The capture of thermal neutrons liberates energy in the form of γ rays. The γ rays are absorbed and produce heat. The distributions of temperatures and thermal stresses caused by this phenomenon in shielding are discussed.

269. Secrest, E. L.

NUCLEAR SHIELDING STUDIES. XII. A QUADRUTURE METHOD FOR COMPUTING NEUTRON DISTRIBUTIONS IN TWO DIMENSIONS. Massachusetts Institute of Technology, Laboratory for Nuclear Science and Engineering,

269. (cont'd) Technical rept. no. 52, N5ori-07818,

16 July 51.

The transport equation governs the scattering and absorption of multiple scattered neutrons during passage through matter. Although the transport equation has not been solved exactly in two dimensions, an iterative method has been developed, which, when combined with a quadrature approximation, allows derivation of approximate solutions of the transport equation.

270. Seetzen, J.

The construction of reactor shielding.

(Technische Hockschule, Hanover).

ATOMWIRTSCHAFT v. 5, p. 230-3,

May 1960. (In German)

In the design and construction of reactor shielding space requirements, weight and cost must be considered. The theoretical fundamentals, the choice of construction materials and the construction technology are discussed with respect to these considerations.

271. Sheehan, W.D.

SHIELDING RESULTS FROM WEAPONS

EFFECTING TESTING. Defense Atomic

Support Agency, Washington, D.C.

Rept. no. NP-10038, p. 38-67.

A discussion is given of measurements for gamma radiation, neutron flux, and shielding experiments during Operations Hardtack and Plumbbob. Graphical and tabulated results are included.

272. Sheehan, W.D. and Allen, F.

PROGRESS ON RADIATION SHIELDING

RESEARCH TANKS. Defense Atomic

Support Agency, Washington, D.C. and

272. (cont'd) Ballistic Research Lab., Aberdeen Proving Ground, Md. Rept. no. NP-10038, p. 293-300.

A description is given of studies made on the transmission of radiation through present combat tanks, and radiation transport characteristics of materials and geometries which are directly involved with the development of a radiological combat vehicle.

273. Sheffield, R. D.
SHIELD SYSTEM OPTIMIZATION--GRADIENT NONLINEAR PROGRAMMING. Convair, Fort Worth, Texas. Rept. no. NARF-57-62T; MR-N-186; Project 6(1-9964), 30 Dec 57, 40p. (Contract AF33(600)-32054)

A procedure is developed for minimizing a nonlinear function of n variables subject to inequality constraints. Under suitable assumptions concerning convexity of the functions, a unique minimum is attainable by the method. The technique is applied to optimization of the total weight of a divided shield system subject to a prescribed dose rate level permitted in the crew compartment and to reactor leakage constraints. The procedure consists of two phases: a dose rate reduction phase followed by an isodose-rate weight reduction phase. Provision is made for satisfying the reactor leakage constraints which are expressed as linear inequalities. The process is iterative and provides a convergence criterion depending upon equality of certain weight-dose coefficient ratios.

274. Shelberg, W. E. and Martin, S. B.
AN EXPLORATORY STUDY OF PROTECTION AGAINST NUCLEAR THERMAL RADIATION WITH SMOKING AND INTUMESCENT PAINTS. Naval Radiological Defense Lab., San Francisco. Rept. no.

274. (cont'd) USNRDL-TR-241, 2 June 58, 43p.

An exploratory laboratory investigation was made on protection against nuclear thermal radiation with thermoshielding and intumescent paints. Thermoshielding paint, a new invention, protects by interposing a rapidly produced, self-generated smoke screen between substrate and radiation. Intumescent paint, commercially available, protects by expanding into a heat-insulating, sponge-like mat. Three experimental, prototype thermoshielding paints were invented, and one intumescent paint, Albi-99, was selected for study. The former consisted of Navy gray paint having tetracene (i.e., 1-guanyl-4-nitrosoaminoguanyltetrazene), nitrosoguanidine or sebaic acid as smoke-generating components. The paints were applied on thin, 3/4-in. diameter, primered copper discs. These were subjected to square-wave and simulated weapon pulse exposures with the Mitchell thermal radiation source, and curves of temperature rise versus time after incidence were determined. Degree of protection was evaluated relative to ordinary Navy gray paint, and was expressed numerically for any particular time after incidence as a percentage (A-B)/(A) x 100, where A represents the temperature rise of the Navy gray paint disc, and B that of the test paint disc. The degree of protection varied with paint and exposure characteristics. The thermoshielding sebacic acid paint protected appreciably (7 to 18%) throughout a range of simulated weapon exposures extending from one of high peak irridance and moderate yield (22 cal/cm^2/sec and 0.2 Mt) to one of low peak irradiance and high yield (5 cal/cm^2/sec and 10 Mt). The explosively decrepitating thermoshielding paints, tetracene and nitrosoguanidine, protected appreciably (13 to 19%) when the simulated weapon pulses delivered the majority of the radiation within a few seconds, but they failed to protect when the majority was delivered during a period of many seconds. Intumescent Albi-99 protected appreciably (7 to 23%) when temperature rises were great enough (about 225 to 250^{o}C) for formation of its relatively slowly produced foam blanket, but it failed completely for large, very rapid temperature rises since immediate decomposition then caused loss of the intumescent property. All paints showed appreciable, sustained protection at exposures corresponding to 22 cal/cm^2/sec peak irradiance and 1-Mt yield: sebacic acid paint, 14%; tetracene and nitrosoguanidine paints, 19%; Albi-99 paint, 23%.

275. SHIELDING SYMPOSIUM PROCEEDINGS.

Shielding Symposium, U.S. Naval Radiological Defense Lab., San Francisco, Calif.

Rept. no. USNRDL-OCDM, R & L no. 110,

31 Oct - 1 Nov 60.

This symposium was organized to provide a review of work going on, the state of the shielding art, and future needs toward development of a broadly applicable shielding technology.

276. Shumway, B. W., Miller, W. G. and Plummer, G. E.

PRELIMINARY MEASUREMENTS OF SHIELDING EFFECTIVENESS OF AN UNDERGROUND FALLOUT SHELTER.

Naval Radiological Defense Lab., San Francisco. Rept. no. NP-10038, p. 179-93.

A study was made of the protection offered by the NRDL-OCDM shelter to give indices of protection attributed to skyshine, lip penetration, and penetration through the roof. A description is given of the shelter which is placed underground and has an L-shaped entrance to have no line-of-sight leakage path for radiation. The protection factors are given according to the location in the shelter.

277. Siemens Schuckertwerke

SHIELD FOR NEUTRON PROBES.

BELGIAN PATENT 572, 809, Nov 29, 1958.

(In French)

Both the neutron probe and its shield are located between the reactor core vessel and the biological shield. In front of the probe is a plate capable of absorbing γ rays; the probe is further protected by a moderating layer, a reflecting layer, and a strong neutron absorber.

278. Sleeper, H. P., Jr.

A CRITICAL REVIEW OF ORNL SHIELD MEASUREMENTS: NEUTRON ATTENUATION.

Oak Ridge National Lab., Tenn. Rept. no. ORNL-436, 21 Dec 49. Decl. with deletions 21 Feb 57, 47p. (Contract W-7405-eng-26)

The purpose of this report is to summarize the neutron data for the most valuable tests of uniform shield where both the shield attenuation and composition are considered well known. Also the results of heavy metal differential neutron measurements are reported, even though the composition of the neutron analyzer is not well known. In these experiments, measurements were made in a uniform "analyzer" shield with and

278. (cont'd) without a heavy metal front slab. The difference is an index of the fast neutron shielding characteristics of the metal.

279. Snyder, W. S. and Neufeld, J.
On the disordering of solids by
heavy corpuscular radiation.
PHYSICAL REVIEW v. 94, n. 3,
p. 760, May 1954.

The general method is outlined for determining the number of vacant lattice sites or interstitial atoms in a monoatomic solid exposed to neutron radiation.

280. Spencer, L. V.
FUTURE NEEDS IN SHIELDING
RESEARCH AND THEORY. National
Bureau of Standards, Washington, D. C.

281. Spiegler, P.
PATCH, AN IBM 709 PROGRAM FOR
CALCULATING GAMMA-RAY BUILDUP
FACTORS FOR MULTI-LAYERED SHIELDS.
Atomics International. Div. of North American
Aviation, Inc., Canoga Park, Calif.
Rept. no. NAA-SR-Memo-5299, 20 May 60,
78p.

PATCH is an IBM 709 Fortran for computing the dose, energy, and energy absorption build-up factor for multi-layered slab geometry shields. The code allows for n regions and six materials with a maximum of seven mesh points per region. The input data consist of the number of regions, the number of materials, their atomic numbers, and the γ energy absorption coefficients of each material. Typical running time for a one-region problem with five integral mesh points is less than 1 min. The running time for a multiregion problem is linearly proportional to the running time of a one - region problem.

282. Spielberg, D. and Duneer, A.

DOSE ATTENUATION BY SOILS AND CONCRETE FOR BROAD, PARALLEL-BEAM NEUTRON SOURCES. Associated Nucleonics, Inc., Garden City, N. Y. Rept. no. AN-108, 1 May 58, 166p. (Contract NBy-13028)

The theory and results of certain neutron shielding studies performed to supply information primarily for underground bomb shelter design are presented. Specifically, for neutron source energies of 0.025 eV, 0.5, 2.5, 7.5, 10, and 14 Mev, and various angles of incidence upon the ground, results are presented that show the total neutron plus secondary gamma dose rates which would be received by individuals located in shelters at various distances below ground. To cover the entire range of possibilities, four basically different soil types were analyzed with water contents ranging from zero to saturation. This range also includes concrete. Also, dose rates were calculated for depths ranging from 30 to 1,000 grams/cm^2. The information presented is considered to be engineering-type data of sufficient accuracy for the intended use. The multi-group theory which was utilized has obvious deficiencies in that certain nuclear scattering characteristics are simplified or ignored. Therefore, the results should not be construed as scientific-type data where comparison with experiment can result in gaining some insight regarding the primary nuclear scattering processes.

283. Spielberg, D.

NEUTRON FLUXES FROM A POINT FISSION SOURCE IN AIR: MOMENTS METHOD CALCULATION. Nuclear Development Corp. of America, White Plains, N. Y. Rept. no. NP-10038, p. 416-23.

The calculations were made using RENUPAK, a neutron moments-method program for the IBM 704. The calculations were based on the point values of the neutron cross sections and angular distributions for nitrogen and oxygen at 215 energies, from 0.009 ev to 18 Mev. The results indicate that the neutron spectra does not vary

283. (cont'd) strongly with distance from the source. This is further illustrated by graphs of the neutron spectra for penetrations of 30, 90, 210, and 400 g/cm^2. A comparison is given between the spectrum at 30 g/cm^2 and the source fission spectra. Neutron doses are shown as a function of distance, computed for a fission source of 1 kiloton (TNT) energy yield.

284. Spielberg, D.

SHIELDING CHARACTERISTICS OF AIR, SOIL, WATER, WOOD AND OTHER COMMON MATERIALS. VOLUME 1. AIR SCATTERING OF GAMMA RAYS AND NEUTRONS. Kidde (Walter) Nuclear Labs., Inc., Garden City, N.Y. Rept. no. WKNL-89 (Vols. 1 and 2), 28 Feb 57 (Vol. 1) 151p. and 25 Mar 57 (Vol. 2), 189p. (Contract DA-44-009-eng-2874(7))

These two volumes were issued separately, but are cataloged as a unit. Basic data and the results of calculations relevant to the problem of reactor shielding with materials readily available at remote reactor sites are presented. The results are presented in graphical form so that shielding needs may be readily evaluated at the reactor site. In essence, this can be called a field-expedient shielding manual. Theory, approximations, and error limits are also given so that the more technical reader is made cognizant of the range of validity of the results. Data are presented on the air scattering of gamma rays and neutrons.

285. Spinney, K. T.

NEUTRON ATTENUATION IN CONCRETE. United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England. Rept. no. AERE-T/R-2507, Feb 55, 15p.

285. (cont'd) Declassified version of AERE-T/R-1617; SWP/P-21. The observed behavior of the penetrating neutrons in a concrete shield is accounted for by a simple theory making use of a calculated 'transport' cross section for fast neutrons. This method is then linked with conventional age theory in an empirical manner so that a good fit is obtained to the observed neutron attenuation at all distances in the shield.

286. Shielding design, theory and practice. Atomic Energy Research Establishment, Harwell, Berks, England. NUCLEAR POWER v. 5, n. 46, p. 134-7, Feb 1960.

Methods of calculating neutrons and gamma penetration into matter and radiation streaming in ducts and voids are reviewed relative to shielding design techniques.

287. Springer, T. and Oberhofer, M. Shielding of neutrons and gamma radiation originating in the shield. (Technische Hochschule, Munich). KERNTECHNIK v. 2, p. 203-5, June 1960. (In German)

For neutron shielding in nuclear engineering operations with Ra-Be or Po-Be neutron sources, neutron generators, or reactors, hydrogen-rich materials (water, paraffin, plastics, oil, tar, etc.), heavy materials (lead, iron, bismuth, concrete, etc), or in many cases combinations of these are used according to the special requirements. The interaction of slow and fast neutrons with these materials is discussed, and formulas are given with which neutron shielding can be estimated without much mathematical effort.

288. Steinberg, H., Heitner, J. and Aronson, R. FANTASIA AND TRIPOD-SHIELDING CODES FOR THE 1103A UNIVAC. Period covered: Apr 58 to Apr 59. TRG, Inc., Syosset, N.Y. Rept. no. WADC-TR-59-443;

288. (cont'd) Project no. 7360, Sep 59, 109p.

(Contract AF33(616)-5187)

Codes were developed for the 1103A Univac (FANTASIA). FANTASIA computes neutron transmission through laminated slab shields and slowing down density within the shields by Monte Carlo methods. TRIPROD is a slowing down code suitable for shielding problems. It is based on the General Electric reactor code VALPROD. Both theory and application of FANTASIA and TRIPROD are discussed. Operating instructions are included

289. Steinberg, H. and Aronson, R.

MONTE CARLO CALCULATIONS OF GAMMA RAY PENETRATION. TRG, Inc., Syosset, N.Y. Rept. no. WADC-TR-59-771; Project no. 7112, Aug 60, 191p. (Contract AF33(616)-6081)

Monte Carlo codes for computing γ-ray transmission through slab and spherical shields were developed. Fractional deviations in transmitted current of above 5% were obtained for shields up to fifteen or twenty mean free paths thick and on iron and aluminum slabs 2 to 10 mean free paths thick at angles of 0, 15, 30, 45, 60, and 75^{o} with the normal. The transmitted energy spectrum of 8 and 10-Mev bremsstrahlung radiation incident on 4 and 6-in. lead slabs was computed.

290. Steinberg, R. and Aronson, R.

High efficiency Monte Carlo calculation of Na^{24} and bremsstrahlung gamma penetration. In Paper 16 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 11p.

TRG's slab γ-code, TRIGR-P, was modified to give improved results for low energies above the cross section minimum in heavy materials. A series of calculations for transmission of Na^{24} radiation (2.75- and 1.38-Mev photons) through aluminum, iron,

290. (cont'd) and lead was run for five to six thickness of from 2 to 12 mean free paths at the higher energy for six incident angles varying from normal to 75° with the normal. There were 192 problems for which both doses and spectra were obtained. The standard deviations for the doses were about 5% for 0, 15, and 30° incidence and about 10% for the larger angles. A few problems gave 15% deviations or more for 60 to 75° on aluminum. The Morton similarity transformation was used to make it possible to obtain results for several thicknesses at once. Spectra from 8- and 10-Mev bremsstrahlung radiation transmitted through 4 and 6 in. of lead were computed and compared with the experimental results.

291. Simpson, K. M., Douglass, C. C. and Stern, H. E.

Crew shield shapes. Lockheed Aircraft Corp.,

Marietta, Ga. In Paper 12 of PAPERS FROM

SEVENTH SEMIANNUAL SHIELDING INFOR-

MATION MEETING, 14-15 OCTOBER 1959.

TID-6302, 8p.

A method for shaping the shielding on the sides of a cylindrical crew compartment is being developed. The shield is designed to achieve minimum weight subject to the requirement of a specified weighted average dose rate along the compartment center line. A machine program was written for a single component of radiation in which attenuation through a single shield material and angular distribution inside are incorporated in the form of general suboutlines.

292. Stern, H. E. and Kam, F. B. K.

DESIGN OF A REACTOR SHIELD.

Oak Ridge National Lab., Tenn.

Rept. no. ORNL-3016, p. 233.

A study of some of the problems inherent in the design of reactor shields for nuclear-powered aircraft was initiated. Calculational methods for obtaining optimum configurations are being explored. Initial attention was focused on γ rays born in a core and utilized a simple model for some approximate calculations.

293. Sunta, C. M.

GAMMA DOSE ATTENUATION

CALCULATOR (Slide Rule).

293. (cont'd) India Atomic Energy Establishment, Trombay. Rept. no. AEET/HP/TH/5, 1 July 60, 40p.

The physical principles underlying the design, condition, and use of a gamma dose attenuation slide are described. The slide rule can be used for shielding calculations for point sources of gamma rays of any energy between 0.3 and 3 Mev and for lead, iron, concrete, and water as shielding material. Graphs and tables of K factors, tenth value layers, and transmission factors with various thicknesses of the shielding materials are included for monoenergetic gamma rays of 0.5 to 3 Mev energy and for a number of complex gamma emitters.

294. Tait, J. H. and Price, J. SOME CALCULATIONS ON THE SHIELDS NECESSARY FOR DIFFERENT SIZES OF EFFLUENT TANKS. Great Britain Atomic Energy Research Establishment, Harwell, Berks, England. Rept. no. AERE-T/M-49, 6 June 51. Decl. 18 Apr 58, 1 p.

Calculations were carried out to determine the most economical way of shielding the effluent in a tank of given size. In the analysis the cylindrical shape was approximated by a spherical one, taking the radius of the latter as equal to half the length of the cylinder. The dose at the outer surface of the effluent tank is calculated, and the effects of multiple scattering in the effluent and shield are assessed.

295. Tati, T. Abandonment of the assumption of the displacement invariance and replacement of the assumption of field by the assumption of particle. Kanazawa Univ., Japan. PROGR. THEORET. PHYS. (KYOTO) v. 24,

295. (cont'd) p. 1-17, July 1960. (In English)

A description is given of a field theory by two parameter spaces in order to separate the two functions fulfilled by the space time, i.e., the representation of the degree of freedom of the dynamical system and the specification of the order of measurements. After the two functions of space time are separated, the assumption of the displacement invariance is abandoned and the momentum-conservation law of finite "range" is assumed. Further, the assumption of the field is replaced by the assumption of the particle. Then the domain of the applicability of theory is extended so as to be able to describe the elementary particles with finite degree of freedom. Some possible observable effects peculiar to the theory of finite degree of freedom are discussed. The difference between the concept of space-time distance in the present field theory and that in the new theory is mentioned.

296. Tanaka, S.

Theory of matter with super light velocity. Kyoto Univ. PROGR. THEORET. PHYS. (KYOTO) v. 24, p. 171-200, July 1960. (In English)

The matter with a super light velocity is treated as one possible model in the course of looking for the physical concept that will essentially govern the future theory of elementary particles. It is investigated according to the canonical quantization to what extent this matter could be reconciled to the usual particle aspect of elementary particles. Further, taking into consideration the interaction of this matter with other particles, it is attempted to quantize this matter so as to derive the Lorentz-invariant but quasi-casual S-matrix and to remove the free state of this matter from possible physical states.

297. Technical Information Service, AEC

REACTOR SHIELDING INFORMATION MEETING, SCHENECTADY, 13-14 MAY 1954. Rept. no. WASH-174, 193p.

Thirty-one papers covering practically all phases of reactor shielding and related studies are included. A chronology of reactor shielding and the present status of shielding theory and microscopic nuclear data as presented in Papers 1 and 2, respectively, lay the foundation for the more specific papers which follow. Specific related theoretical computations and experimental measurements discussed include: neutron and γ attenuation in Hanford shielding materials; neutron and γ transmission through air slots; penetration of 6-Mev γ rays in H_2O, Fe, and Pb; moments method calcula-

297. (cont'd) tion of neutron penetration through light materials; stochastic calculations of γ diffusion; use of Boral in shielding; dose measurements in air and H_2O; slowing down, attenuation, and slant penetration of neutrons in H_2O; γ doses due to irregular geometry in slab shields; flux characteristics of extended sources; secondary γ production; broad-beam γ attenuations; thermal neutron flux calculation for EBR; survey of the BNL shielding program; and Monte Carlo calculations of neutron penetration in H_2O. A book-type subject index has been prepared to facilitate the locating of the information.

298. Terrell, C. W.

RADIATION STREAMING IN SHELTER ENTRANCEWAYS. Illinois Institute of Technology, Chicago. Armour Research Foundation. p. 266-92.

A description is given of a series of analytical and experimental efforts directed toward the determination of gamma ray and neutron distributions in ducts and entranceways. The albedo recipe is applied but with considerable modification and rigor. The agreement of the albedo model and experiment is rather good for gamma rays and poor for neutrons.

299. Thara, C.

On the G_A/G_V ratio in β-decay. Kyoto Univ. PROGR. THEORET. PHYS. (KYOTO) v. 24, p. 211-19, July 1960. (In English)

The ratio of the observed axial vector coupling constant to the bare one in β decay, G_A/G_A^o, is calculated from the composite model for the pion, and the results are tabulated for scalar, tensor, and pseudoscalar coupling.

300. Thoraeus, R.

Iron-ore concrete in radiation protection. Institute of Radiophysics, Stockholm. ACTA RADIOL. v. 54,

300. (cont'd) p. 410-16, Nov 1960. (In English)

The use of iron-ore concrete for the production of protective barriers against high-energy radiation is described. The ingredients and composition are specified and the conditions for the production of a high-grade material are discussed. The protective power of the material is studied on the basis of the attenuation properties and the broad-beam factor.

301. Thorne, C. P.

Concrete properties relevant to reactor

shield behavior. University of Sydney.

J. AM. CONCRETE INST. v. 32, p. 1491-

1508, May 1961.

Available information on the factors affecting the properties of concrete relevant to the determination of the stresses in a reactor shield is examined in detail, and the need for further investigation of several of these properties is noted. As a result of this examination it is possible to state the type of concrete best suited to the particular requirements of reactor shielding. The effect of partial drying of the shield on the distribution of temperature and unrestrained strain is examined theoretically and it is shown that the distribution of moisture content in the shield is of major importance.

302. Tihanyi, L.

Methods for computing the thickness of

shields against gamma-rays. Highway and

Railroad Planning Enterprise, Budapest.

ENERGIA ÉS ATOMTECH. v. 12, p. 7-12,

Jan 1959. (In Hungarian)

A graphical method for computing the lead shield thickness to be used with 25-gamma active isotopes is presented. The equivalent thicknesses for iron, concrete, aluminum, or water shields are given. Shortcomings of the method are discussed.

303. Tiller, H. J., et al

RESULTS OF EXPERIMENTAL INVESTI-

GATION ON GAMMA TRANSPORT AND

303. (cont'd) SHIELDING PROBLEMS. Army Chemical Corps Nuclear Defense Lab., Army Chemical Center, Md. p. 238-50.

The work presented is part of a shielding program to study the protection by structures and fortification against residual radiation from Co^{60} and Cs^{137}. Results are discussed for investigations of open field build-up and dose rates above a contaminated plane, ground penetrating radiation in a cylindrical foxhole, and dose rates in a blockhouse from roof contamination.

304. Tobias, C. A. RADIATION HAZARDS IN SPACE FLIGHT. University of California. Radiation Laboratory, Berkeley, Calif. Rept. no. UCRL-8115, Jan 58. (Contract W-7405-eng-48)

The knowledge pertinent to cosmic ray dose evaluation is reviewed for conditions of flight in space in the vicinity of the earth but away from the influence of its atmosphere or magnetic field. Certain properties of the heavy nuclei are also discussed as well as some of the available information on their actual biological effects. Finally the statistical nature of the hazard from cosmic radiations and the limitations of space flight due to such rays are presented.

305. Toll, E. Determination of neutron and secondary photon spectra from a thick laminated shield with spherical symmetry. Naval Air Development Center, Johnsville, Pa. In Paper 8 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION

305. (cont'd) MEETING, 14-15 OCTOBER 1959.

TID-6302, 10p.

A Monte Carlo program, evolved for the photon problem, is being applied by using photon source data from neutron fluxes calculated by a combination of age and diffusion theory. A modification of the Carlson S_n method and a direct analytical method are described which were considered for solving the integro-differential transport equation in the neutron spectra determinations. An analytical solution was developed for the case of isotropic scattering in the laboratory system.

306. Tomoeda, S., et al

Gamma-ray penetration into the compartments of a light aircraft carrier. Naval Radiological Defense Lab., San Francisco. In Paper 18 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 21p.

The penetration of γ rays into compartments of an aircraft carrier from three radioisotopes were measured. Co^{60} (1.25 Mev), Cs^{137} (0.66 Mev), and Ir^{192} (0.38 Mev) sources were exposed at numerous points on the flight deck, and Landverk air-wall equivalent dosimeters were used to measure dose in two general areas of the ship and at three different deck levels. Average dose rates and transmission factors were calculated by integration of the data to represent the case of uniform contamination of 1 curie per square foot. Point source data were compared with data taken using a uniformly distributed source. Effective shielding thicknesses of a ship's structure were determined from experimental data and compared with known structural thicknesses. Energy dependence of the shielding effectiveness can be adequately determined by using attenuation coefficients and build-up factors for the energies involved.

307. Tomoeda, S., Hastings, M.B. and Shumway, B.W.

SOURCE GEOMETRY EFFECTS ON GAMMA-RAY PENETRATION INTO A COMPARTMENTED STRUCTURE. Naval Radiological Defense Lab., San Francisco.

307. (cont'd) Rept. no. NP-10038, p. 223-30.

Dose rates were measured in the various compartments using air-wall equivalent dosimeters with the source at various distances from the box. The results are presented as geometry factors as a function of source-to-detector distance. The geometry factor is defined here as the ratio of the measured dose rate to the unattenuated primary dose rate calculated for a given point. The need for this type of information arises in working out prediction methods for dose rates inside complex structures, such as ships and buildings caused by volume distributed contaminations.

308. Troubetzkoy, E. and Goldstein, H.

A COMPILATION OF INFORMATION ON

GAMMA-RAY SPECTRA RESULTING

FROM THERMAL-NEUTRON CAPTURE.

Nuclear Development Corp. of America.

Rept. no. ORNL-2904, 18 Jan 61.

(Contract W-7405-eng-26, Subcontract

no. 1216)

This report consists of a compilation of thermal-neutron capture gamma-ray spectra of various nuclei which are useful for reactor shielding applications. All the data are presented in tabular form for seven energy intervals from 0 to above 9 Mev. In addition, figures illustrating the appearance of the spectra are given, including continuous differential spectra for 50 nuclei and line spectra for 15 nuclei. The line spectra consist of vertical lines having heights proportional to intensities.

309. Trout, E.D., Kelley, J.P. and Lucas, A.C.

Conventional building materials as protective

radiation barriers. General Electric Co.,

Milwaukee. RADIOLOGY v. 76, p. 237-44,

Feb 1961.

A study was made of the radiation shielding provided by various conventional building materials. An x-ray unit, provided with a diaphragm that could be adjusted for square or rectangular fields, was used as the source. Structural panels of 8 typical wall materials were tested. These panels ranged in thickness from a thin gypsum plaster

309. (cont'd) wall to a thick cinder block and Portland cement mortar wall. Radiographs were made of representative areas of each wall section. The transmission data were converted to lead equivalents and also to concrete equivalents. A schematic sketch is included of a typical radiographic room presenting typical shielding situations. Data are tabulated on the thickness of lead which would have to be added to a particular wall to provide adequate shielding when using the various types of construction. It is concluded that the shielding effects of masonry walls is sufficient to provide useful barriers for many applications. Walls of conventional building materials should be considered for shielding calculations even though they may not provide all the shielding necessary in a given wall. A considerable reduction in the amount of lead required will usually be effected.

310. Trubey, D. K.

AN ESTIMATION OF PHOTONEUTRONS

FROM CARBON-13 IN AN OIL SHIELD.

Oak Ridge National Lab., Tenn. Rept.

no. ORNL-2200, 29 Aug 58, 19p.

(Contract W-7405-eng-26)

An expanded version of data previously published in ORNL-2081, p. 168 (Sep 10, 1956). Measurements of thermal-neutron flux in the presence of high-energy gamma rays in oil indicate that photoneutrons comprise a large fraction of the total thermal-neutron flux at large distances from the neutron source. This is attributed chiefly to the (γ, n) reaction in C^{13}. Measurements at the Lid Tank Shielding Facility in oil with and without a bismuth slab to reduce the gamma rays, and consequently the photoneutrons which they produced, agreed with calculated fluxes.

311. Tsvetave, N. E.

Shape of the absorption curve for beta

rays in aluminum. ATOMNAYA ENERG.

v. 9, p. 507-8, Dec 1960. (In Russian)

The absorption curves for simple β emission spectra with E_o = 200 and 600 kev and atomic β emission number Z = 4, 20, and 80 were calculated by graphical integration. The half-absorption layers found by calculating the initial ($D_{1/2\ init}$) and mean ($d_{1/2\ mean}$) show a considerable difference. Moreover, $D_{1/2\ init}$ is a function of the β emitter atomic number. The experimental values for half-absorption layers ($D_{1/2\ init}$) for S^{35}, Ca^{45}, Co^{60}, and Sr^{90} correlated to the obtained data are in good agreement except for Sr^{90}.

312. United Kingdom Atomic Energy
Authority. Industrial Group H. Q.,
Risley, Lancs, England.
RADIATION SHIELDING. INFORMATION
BIBLIOGRAPHY. Rept. no. IGRL-IB/R-30,
Sep 57, 14p.

This selection of references is primarily for information on reactor shielding and its design. Sources searched were U. K., Canadian, and U. S. report abstracts, Nuclear Science Abstracts, Engineering Index (1950-1955), and Industrial Arts Index (1956).

313. Unterberg, W.
THIN BODY TEMPERATURE TRANSIENTS
DUE TO GAMMA HEATING IN AEROSPACE. Rocketdyne,
North American Aviation, Inc. , Canoga
Park, Calif. (Presented at Space-Nuclear
Conference, 3-5 May 1961, Gatlinburg, Tennessee)
Paper no. 1759-61. New York, American
Rocket Society, 26p.

Components of nuclear propulsion and power systems are subject to gamma heating caused by the reactor. These components may be cooled by convection and thermal radiation to the aerospace environment and other parts of the spectrum, such as plates, tubes, and shells with small internal temperature and heat generation rate gradients, are evaluated analytically for simple-step and certain time-dependent heating and cooling inputs. For time- and temperature-varying inputs, numerical methods are outlined. Maximum body temperatures, found without solving the basic differential equation, permit shortcuts in design calculations. Thermal radiation is a significant cooling mode. Also discussed are temperature-dependent inputs, and adaption of the methods presented to extended shapes and complex geometries.

314. Vacca, G.

Protection against γ radiation. Graphic determination of shields. (Commissariat à l'Énergie Atomique, Paris). J. RADIOL. ET ÉLECTROL. v. 39, p. 498-504, 1958. (In French)

A set of abaci with which the thickness of γ radiation shields may be determined are presented. For monoenergetic radiation, the result may be read directly without any calculation. In the case of multiple gamma energies, the thickness required can be estimated with very simple calculations. The principles of the calculation are briefly described, and the method of using the abaci is illustrated.

315. van Ammers, Ir. H.

State of development in the range of reactor shielding. II. Theory and practice. ATOMENERGIE v. 2, p. 123-9, Aug 1960. (In Dutch)

In this second part of the article on reactor shielding, neutron shielding is first discussed. Then safety shielding is considered. A brief survey on radiation detection instruments used for examination of shielding is given.

316. van Ammers, Ir. H.

State of development in the range of reactor shielding. I. Theory and practice. ATOMENERGIE v. 2, p. 107-15, July 1960. (In Dutch)

In the design and calculation for the shielding of a nuclear reactor, one encounters problems of a practical and theoretical nature. As in most practical cases one has to deal with shielding against both gamma rays and neutrons; this subject is discussed in detail first. It is followed by a survey of results attained in recent years. The article ends with a description of radiation detection instruments for shielding research.

317. van Roosenbeek, E. and Grimm, J.H.

Wedge filters: Their construction and use with the 22 Mevp betatron. Univ. of Texas M. D. Anderson Hospital and Tumor Inst., Houston. AM. J. ROENTGENOL., RADIUM THERAPY NUCLEAR MED. v. 84, p. 926-32, May 1961.

The physical aspects and construction details of wedge filters by a calculative method are presented for 22 Mevp roentgen rays. The calculative method produces isodose curves of wedged fields that are in good agreement with measured isodose curves. Their clinical application and volume distributions in 22 Mevp therapy are given.

318. Vaidya, P. C. and Shah, K. B.

A relativistic model for a shell of flowing radiation in a homogeneous universe. Gujarat Univ., Ahmedabad, India. PROGR. THEORET. PHYS. (KYOTO) v. 24, p. 111-25, July 1960. (In English)

Rigorous solutions of Einstein's field equations of general relativity are presented. They are found to represent the gravitational field within a shell of radiation flowing outward in a homogeneous universe. These models suggest that the effect of complete conversion of matter into radiation appears to decrease the curvature of the universe.

319. Vroom, D. W.

PRELIMINARY MSRE GAMMA RAY SOURCE AND BIOLOGICAL SHIELDING SURVEY. Oak Ridge National Lab., Tenn. Rept. no. CF-61-4-97, 28 Apr 61, 41p.

319. (cont'd) The major gamma ray sources and some of the resultant biological shielding requirements were evaluated at both full power operation and following shut down for the Molten Salt Reactor Experiment. The sources include gamma rays due to the fission events in the reactor and those associated with activation of materials. Shielding estimates were made for the reactor system cells and for some maintenance facilities.

320. V'yugov, P. N., et al

The attenuation of gamma rays by concrete and by certain soils.

ATOMNAYA ENERG. v. 10, p. 76-9, Jan 1961. (In Russian)

The economic aspects of soil and sand shielding are discussed. Tabulated data for 1-m^3 of various shielding materials for a 2-Bev linear accelerator indicate that large savings result from replacing concrete by soil and sand.

321. Wallace, R. R., Vinson, J. R. and Kornhauser, M.

EFFECTS OF HYPERVELOCITY PARTICLES ON SHIELDED STRUCTURES.

General Electric Co., Missile and Space Vehicle Dept., Philadelphia, Pa. (Presented at the ARS Lifting Re-entry Vehicles; Structures, Materials and Design Conference, Palm Springs, Calif., 4-6 Apr 1961.) American Rocket Society, Inc., New York, N. Y. rept. no. 1683-61.

A test program was conducted in which over 300 hypervelocity projectiles were fired at structural walls protected by single shields. Projectile material, projectile size, projectile velocity, shield material, shield thickness and shield to wall spacing were the variables investigated. Some new concepts which should permit more scientific hardening and a more complete understanding of particle impact are discussed. An extensive statistical analysis of variance for the five phases of the test program and a

321. (cont'd) three-step breakdown analysis involving each of the above parameters were accomplished. Design criteria and trends, based upon pertinent results and the statistical analyses, are presented. It was found that in these structures over 50% of the external structural weight for single-walled particle protection can be saved by proper design and use of meteor bumpers without sacrifice of safety.

322. Wallner, L. E. and Kaufman, H. R.

RADIATION SHIELDING FOR MANNED

SPACE FLIGHT. Rept. U. S., NASA TN

D-681, July 61, 45p.

Calculation of radiation shield weights required for space vehicles. Cosmic radiation, solar flares, the Van Allen belts, and radiation from a reactor are considered for both short and long travel times. For the long-term mission to Mars, cosmic and solar-flare radiation may necessitate shield weights of 100,000 lbs. Increasing the radiation dose several hundred percent above present acceptable levels does not eliminate the need for heavy shields. Shields required against radiation from the reactor and the Van Allen belts are a level of magnitude less, except for slow traversal of the earth's radiation belts. The use of a common shield against all the various hazards would effect a considerable weight saving. For long-term missions - e.g., to Mars assembly of the vehicle in space probably will be necessary. It is possible that a partial body shield affording adequate protection for short exposure times could be designed with a weight penalty as low as 100 lbs. per man.

323. Watts, E. A.

Lead suspension for low-energy radiation

shielding. Columbia Univ., New York.

HEALTH PHYS. v. 4, p. 68-9, Oct 1960.

A lead suspension was devised which may be applied easily by brush or spray-gun to provide additional shielding for existing intermediate and low-energy x-ray tube housings, darkrooms, and radiation enclosures. The suspension forms a continuous surface, conforms to existing contours, and is capable of being finished to match existing decor.

324. Wells, M. B.

Monte Carlo calculations of fast-neutron

energy spectra in air. Convair, Fort Worth, Tex.

In Paper 11 of PAPERS FROM SEVENTH

324. (cont'd) SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 18p.

Results are presented of Monte Carlo air-scattering calculations for source-neutron energies upon the threshold energy of inelastic scattering. Energy and angular distributions of the scattered neutron flux and angular distributions of the scattered dose rate are given for point isotropic monoenergetic sources at source-detector separate distances to 200 ft. The results are compared with calculations by Holland and Richards and by Zerby to evaluate the effect of inelastic scattering on the scattered energy spectrum.

325. Wells, M. B.

A MONTE CARLO CALCULATION OF GAMMA-RAY AND FAST-NEUTRON SCATTERING IN AIR. Convair, Fort Worth, Tex. Rept. no. NP-10038, p. 308-95.

The calculations were made for source-detector separation distances up to 1500 yards to provide information on the radiation field resulting from a nuclear weapon burst. The calculated fluxes and dose rates were multiplied by $r\pi R^2$, where R is the source-detector separation distance, and the results normalized to a point isotropic source emitting one particle per second. The energy and angular distributions are tabulated for dose rates for gamma source energies of 0.5 to 6 Mev, and for fast neutron source energies of 0.75 Mev, and for fast neutron source energies of 0.75 Mev through fission. The average standard deviation for all problems was 18% of the calculated scattered flux. The standard deviation exceeds 20% in only 20% of the problems. The accuracy of the energy spectra and angular distributions should be within ± 30%.

326. Wells, M. B.

STUDIES IN SHIELDING-III. MONTE CARLO CALCULATIONS OF NEUTRON SCATTERING IN AIR. Convair, Fort Worth, Texas. Rept. no. NARF-57-49-T;

326. (cont'd) Project 6(1-9964); FZK-9-120, 4 Nov 57, 47p. (Contract AF-33(600)-32054)

A Monte Carlo method is described for calculating the angular distribution and energy spectrum of air-scattered neutrons. The method was used to solve the integrals that represent the contribution to the total-scattered flux from each order of scattering. Results of the calculations giving the scattered flux from a 3.25-Mev point monodirectional source are presented in both graphical and tabular form for various source angles and separation distances at 10,000-ft. density altitude.

327. Welsher, R. A. G. Remote handling. IV. Shielding systems. A. E. R. E., Harwell. NUCLEAR ENG. v. 2, p. 427-30, Oct 1957.

Various types of materials for shielding against high-energy radiation are considered.

328. Wethington, J. A., Jr., Karam, R. A. and Bisselle, C. A. GAMMA RAY SCATTERING. Florida Univ., Gainesville. Final technical progress rept. for 18 Jan 59 - 18 June 61, 19p. (Contract N0as 59-6015-c)

Studies were made of gamma scattering to see if any effects could be observed when solid shields were replaced with fine powders. Aluminum, copper, and lead powders of various sizes were used as shields in both narrow and broad beam geometry, and these powders were compared with the corresponding solid element. No differences were found. Experiments confirmed theoretical predictions that the addition of electrons to the surface of a metal should cause no observable change in the mass attenuation coefficient of the material. Pseudo mass attenuation coefficients for polyethylene and lead were determined in a broad beam geometry. At certain thicknesses polyethylene attenuated gamma rays in a more efficient manner than lead of equivalent surface density. Final results obtained with the true photon spectra showed that shields made of polyethylene and lead were more efficient than lead shields. It was shown experimentally that it was not necessary to convert pulse-height data into photon spectra in order to obtain relative dose rates from the various shields.

329. Wheeler, D. M. and Bostick, L. H.

MILITARY FIELD EXPEDIENT

SHIELDING EXPERIMENT. Convair

Nuclear Research and Development Lab.,

Fort Worth, Texas. Rept. no. FZK-122,

18 Oct 60, 233p. (Contract DA-44-192-

eng-13)

Experiments were performed to obtain data for use in applying field materials to shield portable nuclear-reactor power plants. Experimental data were obtained on neutron and gamma-ray attenuation and on neutron activation for representative field materials and on air scattering of radiation. Field materials tested were dry sand, wet sand, clay, limestone, water, a soil mixture, and pine wood. Results of the experiments showed that a given thickness afforded more fast-neutron attenuation than gamma-ray attenuation. Secondary-gamma-ray production in expedient materials is an important consideration dependent on the water (hydrogen) content in the material. Significant amounts of radioactivity induced in expedient materials can be caused by trace elements. Activation tests of material samples from anticipated sites are desirable. Secondary-gamma rays produced in air are determined primarily by the type of reactor and shield and are a significant fraction of the air scattered gamma-ray dose rate.

330. White, W. E.

HIGH ENERGY GAMMA RAY PENETRATION

IN LEAD. Fairchild Engine and Airplane

Corp. NEPA Div., Oak Ridge, Tenn.

Rept. no. NEPA 1324 STR, 22 Mar 50.

A theoretical paper on the penetration of lead by 20 mc^2 gamma rays.

331. Whiton, J. C. and Burrell, M. O.

MONTE CARLO INVESTIGATION OF

NEUTRON CAPTURES IN BORATED IRON

SLABS. Lockheed Nuclear Products,

331. (cont'd) Marietta, Ga.

p. 453-67.

The program treats an incident beam of monoenergetic neutrons from either an isotropic or a monodirectional source and computes boron and iron capture distribution, and albedo and transmission characteristics. The investigation was carried out to determine the secondary gamma dose in one- and three-inch iron slabs with a boron content of zero to seven per cent by weight, at incident energies of from 0.1 to 3500 ev.

332. Wood, D. E.

NEUTRON ATTENUATION OF MAGNETITE CONCRETE HEATED TO 100 C. General Electric Co. Hanford Atomic Products Operation, Richland, Wash. Rept. no. HW-53395, 29 Oct 57. Decl. 25 Feb 58, 13p. (Contract W-31-109-eng-52)

Neutron attenuation of Magnetite concrete heated to 100°C was measured in the DR shield test well. Gold and S foil detectors were used to determine fast, epithermal, and thermal fluxes at intervals through 48 in. of concrete. The fast neutron relaxation length is 9.52 cm, compared to 8.75 cm for the unheated concrete, a 9% increase as a result of this heating. Thermal and epithermal attenuation curves show little indication that water loss has been sufficient to significantly change intermediate energy neutron leakage. Side to side traverses indicate the possibility that streaming up the edges of the well is affecting the measurements near the outside of the well. This does not affect the fast neutron relaxation length which is determined in the first 32 in. of concrete. However, the thermal and epithermal traverses would normally be used to confirm the relaxation length in the outer part of the well where spectral equilibrium should be established. This procedure is not valid if serious streaming is present.

333. Wood, D. E. and Peterson, E. G.

TRANSVERSE STREAMING TESTS IN THE DR SHIELD TEST WELLS. General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

333. (cont'd) Rept. no. HW-56593, 30 June 58, 8p.

(Contract W-31-109-eng-52)

Past experiments on neutron attenuation of various materials conducted in the vertical shield test wells on top of the DR reactor has led to the question of streaming out of the well or up the edges of the wells. Horizontal traverses were taken in the DR vertical shield test wells to investigate the problem of neutron streaming. Both fast and thermal neutron traverses were taken to within three inches of the edge of the well. A temperature traverse showed that the concrete in the well was quite warm.

334. Wood, D. E. and Peterson, E. G.

NEUTRON ATTENUATION IN MAGNETITE CONCRETE HEATED TO 200°C. General Electric Co. Hanford Atomic Products Operation, Richland, Wash. Rept. no. HW-55372, 25 Mar 58, 15p. (Contract W-31-109-eng-52)

The relaxation length for fast neutron in magnetite concrete heated to 200°C is 9.74 cm, compared to 9.52 cm after heating to 100°C, and 8.75 cm before heating. The largest change in relaxation length occurred as a result of heating to 100°C. It appears that at or below 100°C, the concrete loses more water than in subsequent heating steps. Attempts to confirm the water loss by weighing the slabs were not accurate enough to be conclusive, although definite weight losses occurred during each heating.

335. Yarmush, D.

Gamma penetration by transmission matrix method. TRG, Inc., Syosset, N. Y. In Paper 17 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 8p.

A code for computing transmission and reflection matrixes for γ rays on slabs was written and debugged. The code first computes an auxiliary matrix giving the properties

335. (cont'd) of infinitely thin slabs by using empirical cross sections and the Klein-Nishina formula. The flux is presented by dividing the energy range into intervals and by using cosine powers for separate expansions in angle over the forward and backward hemispheres. An option allows for other methods of angular representation, e.g., the half-range Legendre polynomials. Preliminary numerical results are given.

336. Yarmush, D., Zell, J. and Aronson, R.

THE TRANSMISSION MATRIX METHOD FOR PENETRATION PROBLEMS. Period covered: Dec 58 - Nov 59. TRG, Inc., Syosset, N.Y. Rept. no. WADC-TR-59-772; Project no. 7112, Aug 60, 299p. (Contract AF33(616)-6081)

A practical machine method for computing transmission and reflection of radiation by laminated slab shields is described, making use of transmission and reflect matrices. These are combined in a way which produces a formal solution of the Boltzmann equation in slab geometry. This result is applied by using representations in terms of energy groups and angular expansions. Two codes for the IBM-704 which together carry out the computation of the transmission and reflection matrices are described and operating instructions are given. Numerical results for gamma rays are presented. The advantages of the transmission matrix method as compared to other methods are discussed. Appendices are included showing how in the P_1 approximation the method gives diffusion theory in a particularly simple way and how it can be applied to other geometries.

337. Zerby, C.D.

Monte Carlo code for the calculation of response functions of gamma-ray scintillation detectors. Oak Ridge National Lab., Tenn. In Paper 7 of PAPERS FROM SEVENTH SEMIANNUAL SHIELDING INFORMATION MEETING, 14-15 OCTOBER 1959. TID-6302, 10p.

A Monte Carlo code was written for the IBM-704 computer for calculations of the γ-ray response functions of NaI and xylene scintillation counters. The counter geometry can be either a right cylinder or a right cylinder with one conical end. The source is restricted to a monoenergetic source of arbitrarily chosen energy of 0.005 to 10.0

337. (cont'd) Mev. The treatment of the primary incident radiation takes into account Compton scattering, pair production, and the photoelectric effect. The secondary bremsstrahlung and annihilation radiation are considered in the case of a NaI crystal. The particular Monte Carlo method used is designed for minimum statistical error in the so-called "Compton tail" of the spectrum.

338. Zirkind, R.

TECHNICAL SUMMARY OF ALL ORNL SUMMER SHIELDING SESSION. (1949)

Oak Ridge National Lab., Tenn. Rept. no. ORNL-503, 19 June 50. Decl 22 Feb 57, 50p. (Contract W-7405-eng-26)

An evaluation was made of the theoretical and experimental aspects of both neutron and gamma reactor shielding. Basic data are reviewed and analytical and numerical methods are discussed. Neutron cross-section data and gamma-ray absorption coefficients of particular interest in shielding studies are included.

SOURCE INDEX

Aeronutronic, Newport Beach, Calif. 21

Air Force Inst. of Tech., Wright-Patterson Air Force Base, Ohio. . . . 30

Aktiebolaget Atomenergi, Stockholm 34, 37

Alco Products, Inc., Schenectady, N. Y.. 210

Argonne National Lab., Lemont, Ill. 86, 120

Army, Washington, D. C.. 218

Army Chemical Corps Nuclear Defense Lab., Army Chemical Center, Md. 77, 303

Army Engineer Research and Development Labs. Fort Belvoir, Va. 128

Army Medical Research Lab., Fort Knox, Ky. 163, 182

Arthur D. Little, Inc., Cambridge, Mass. 91, 92

Associated Lead Manufacturers Ltd. 14

Associated Nucleonics, Inc., Garden City, N. Y. 282

Atomic Energy of Canada Ltd., Chalk River, Ont.. 264

Atomic Power Development Associates, Inc., Detroit 146, 219

Atomics International. Div. of North American. Aviation, Inc., Canoga Park, Calif. . 12, 13, 83, 165, 281, 111

Avco-Everett Research Laboratory, Everett, Mass. 193

Ballistic Research Labs., Aberdeen Proving Ground, Md. 5

Battelle Memorial Inst., Columbus, Ohio 93

Boeing Airplane Co., Seattle 205, 235

California Univ., Berkeley 172

California Univ., Los Angeles. Atomic Energy Project 189

Centre d' études Nucleaires, Fontenay-aux-Roses, France 24

Commissariat à l' Énergie Atomique, Paris, France. 25, 26

Convair, Fort Worth, Texas . . . 19, 41, 69, 75, 84, 87, 88, 102, 104, 159
161, 162, 174, 207, 214, 237, 243, 324, 325, 326, 329

Defense Atomic Support Agency, Washington, D. C. 271, 272

Defence Research Chemical Lab., Ottawa 59

Division of Reactor Dev., AEC 76

Fairchild Engine and Airplane Corp. 330

Florida, Univ., Gainsville 46, 206, 328

Ford Motor Co., Newport Beach, Calif.. 20

France. Commissariat à l' Energie Atomique. 79
Centre d' Etudes Nucleaires, Saclay

General Electric Co., Milwaukee 309

General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinnati. . 89, 107
122, 123, 126

General Electric Co., Aircraft Nuclear Propulsion Dept. . . 48, 54, 125, 198
Evendale, Ohio

General Electric Co. Hanford Atomic Products 47, 223
Operations, Richland, Wash. 246, 247, 332, 333, 334

Georgia Inst. of Tech., Atlanta. School of Civil Engineering. 267

Gt. Brit., Armament Research and Development 220
Establishment, Fort Halstead, Kent, England

Gt. Brit. Atomic Energy Research Establishment, ... 38, 139, 294
Harwell, Berks, England

Gt. Brit. Building Research Station, Garston, ... 118
Watford, Herts, England

Gt. Brit. Dept. of Atomic Energy. Industrial 244
Group H. Q., Risley, Lancs, England

Illinois Inst. of Tech., Chicago. Armour Research Foundation. 298

India. Atomic Energy Establishment, Trombay. 293

Institut fur Medizinische Physik and Biophysik, ... 85
Gottingen, Germany

Inst. of Materials and Tech., Prague 103

Iowa. State Univ., Iowa City 226

Japan. Atomic Energy Research Inst., Tokyo 72

Kaiser Engineers Div., Henry J. Kaiser Co., ... 82
Oakland, California

Knolls Atomic Power Lab., Schenectady, N.Y. 108, 181, 256

Lockheed Aircraft Corp., Missile Systems Div. 195
Sunnyvale, California

Lockheed Nuclear Products, Marietta, Ga.. . . . 6, 50, 134, 135, 291, 331

Los Alamos Scientific Lab., New Mexico 97, 105

Massachusetts Institute of Technology, Laboratory for 71, 269
Nuclear Sciences and Engineering, Cambridge, Mass.

Michigan. Univ., Ann Arbor. Engineering Research Inst.. 73

NASA, George C. Marshall Space Flight Center, ... 160
Huntsville, Alabama

National Aeronautics and Space Administration 9

National Bureau of Standards, Washington, D.C. 90, 280

Newport News Shipbuilding and Dry Dock Co., Va. 203

North American Aviation, Inc. 45

Naval Air Development Center, Johnsville, Pa. 305

Naval Air Material Center 49

Naval Civil Engineering Lab., Port Hueneme, Calif. . . 57, 58, 78, 190, 211

Naval Radiological Defense Lab., San Francisco 124, 168, 183, 184
274, 275, 276, 307

Nuclear Development Corp. of America 2, 114, 156, 252, 254, 283
White Plains, N. Y.

Nuclear Shielding Supplies and Service, Inc. 237
White Plains, N. Y.

Oak Ridge National Lab., Tenn. . . . 16, 31, 32, 34, 35, 36, 39, 51, 55, 56
60, 61, 62, 63, 64, 65, 66, 67, 68, 116, 144, 157
187, 201, 213, 224, 233, 245, 278, 292, 310, 319, 337, 338

Office of Civil and Defense Mobilization, Battle Creek, Mich. . . 98, 119, 265

Ontario Cancer Inst., Toronto 27

Osaka Univ. Japan 10

Radiation Effects Information Center, Columbus, Ohio 258A

Rand Corp., Santa Monica, Calif. 129

Rocketdyne, North American Aviation, Inc., ... 313
Canoga Park, Calif.

Technical Information Service, AEC 255, 297

Technical Operations, Inc., Burlington, Mass. 23

TRG, Inc., Syosset, N. Y. 170, 171, 288, 289, 290, 336, 335

United Kingdom Atomic Energy Authority, Industrial 312
Group H. Q., Risley, Lancs, England

United Kingdom Atomic Energy Authority. Research. . . 1, 28, 29, 99, 285
Group. Atomic Energy Research Establishment,
Harwell, Berks, England

Universität, Tübingen, Germany. 42

University of California, Los Alamos Scientific Lab.. 117

University of California, Radiation Lab., Berkeley, Calif.. . . . . 133, 304

U. S. Army Signal Research and Development Lab., Fort 147
Monmouth, New Jersey

U. S. Atomic Energy Commission 17, 18

Utah Univ., Salt Lake City 52

Univ. of Rochester, N. Y. 53

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh 208

Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh. . . 7, 8

Wright Air Development Center. Air Force Nuclear Engineering Test Facility. Wright-Patterson AFB, Ohio 96

Wright Air Development Center, ANP Advisory Committee for Nuclear Measurements and Standards 143

SUBJECT INDEX

Air
Shielding properties 284

Air-water systems
Gamma and neutron distribution 68, 174

Aircraft
Divided shield concept 191
Partial shields 207, 243
Shield design 292, 291, 293
Shield development 69
Shielding materials 101
Shielding optimization 214
Shield studies 35
Shielding program 122, 123

Aircraft shields
Scattered and secondary radiation inside 48

Aluminum
Beta absorption 311
Gamma scattering 328
Gamma penetration. 290
Gamma shielding properties 99
Neutron attenuation 45
Neutron elastic scattering 133
Shielding properties for space craft 20
Shielding thickness calculations 302

Aluminum-boron systems
Gamma penetration 297
Shielding properties 93

Aluminum-boron carbide systems
Radiation attenuation 51

Aluminum-iron systems
Gamma buildup factors 166

Aluminum-lead systems
Gamma buildup factors 166

Armored vehicles
Radiation doses inside from nuclear weapons 235

Beryllium
Gamma and neutron distribution 63, 66
Neutron attenuation, thermal 56
Neutron elastic scattering 133

Beryllium-iron alloys
Gamma and neutron dose rates 55

Beryllium-iron-systems
Gamma and neutron dose distributions beyond 61

Beryllium stainless steel systems
Gamma and neutron dose rates 55

Beta radiation
Absorption in aluminum 311
Shielding 227

Bismuth
Shielding properties 287

Boral
Neutron attenuation 24

Boron
Gamma and neutron shielding properties 241

Boron carbide-iron systems
Gamma and neutron distribution 66

Boron carbides
Gamma and neutron distribution 63, 64
Gamma attenuation 43
Shielding properties 81, 96, 241

Boron-carbon systems
Neutron absorption 221

Boron-carbon-graphite
Preparation of shielding materials 196

Boron-graphite systems
Shielding properties 93

Boron-iron systems
Neutron capture in slabs 331

Building materials
Shielding properties of conventional 309

Bulk shielding reactor
Neutron fluxes 33

Carbon
Neutron elastic scattering 133

Ceramic materials
Shielding properties 250

Clay
Gamma shielding properties 128

Computers
Code for gamma attenuation in cylindrical and spherical geometries . . . 83
Gamma penetration point source program. 89

Concrete
Economics as shielding 176, 177
Gamma absorption coefficient 74
Gamma attenuation 11, 12, 47, 236, 320
Gamma attenuation for slab shields 78
Gamma shielding properties 131
Gamma transmission 27, 258
Heating and radioactive emission effects on shields 178
Neutron absorption coefficients 209
Neutron attenuation 25, 34, 45, 47, 246, 247, 282, 285, 332, 334
Nuclear properties 36
Physical properties 36, 267
Shield design 79, 82
Shielding properties 80, 118, 121, 149, 150, 151, 152, 153
154, 179, 215, 219, 224, 234, 287, 300, 301
Shielding thickness calculations 302
Survey for radiation shielding 211
Thermal properties of shielding 142

Concrete-iron systems
Neutron fluxes and gamma dose rates 105

Concrete-steel-water systems
Neutron attenuation 264

Copper
Neutron elastic scattering 133
Gamma scattering 328

Cosmic protons
Effects on manned space flight 9
Shielding materials 21

Cosmic radiation
Space vehicle shield design 22
Tissue and water ionization by 100

Diffusion theory
Validity for shielding analysis 2

Electrons
Shielding materials 159, 160, 161

Ethylene polymers
Neutron transmission through borated 5
Shielding properties 14

Fir
Gamma shielding properties 128

Gamma radiation
Absorption coefficients for various media 248
Attenuation calculator 293
Attenuation in concrete 13, 34, 47
Attenuation in lead 39, 62
Attenuation in shielding materials 63, 175A, 180, 202, 297
Calculation in shielding 1
Calculation of air-scattered transfer functions 134
Capture from infinite slab 137
Energy spectra 19
Penetration in soil 59
Distribution in iron and water 158
Penetration in lead and water 39
Penetration through a multilayer shield 29
Penetration in composite slabs 156
Penetration, Monte Carlo calculations 289
Penetration in shields 113
Scattering from various materials 136
Scattering in air 325
Scattering in shield labyrinth 10
Shield thickness determination 302, 314
Shielding 12, 188
Shielding of 1.5 Mev 11
Shielding calculations 238
Shielding for in-pile experiments 208
Shielding for irradiation facility 225
Shielding materials 128
Spectral measurements 104
Transmission through slabs of various materials 254
Transport, effect of air-sand interface. 172
Transport problem 16

Gamma sources
Shielding requirements 27

GASLIT
Program for calculating gamma dose rates in shields 1

Glass
Shielding properties 15
Neutron penetration 94

Granite
Gamma shielding properties 128

Graphite
Neutron absorption cross sections 110

Handbooks
Activation for aircraft designers 84

Iron
Gamma absorption coefficient 74
Gamma attenuation 63, 64, 66, 115, 175A, 297
Gamma distribution 158
Gamma penetration 290
Gamma shielding properties 99, 121
Gamma transmission 258
Neutron attenuation 44, 45, 56, 63, 64, 66
Neutron transport through slabs 50
Shielding thickness calculations 302
Shielding properties 287

Iron-lead systems
Design of shielded containers 103
Gamma buildup factors. 166

Iron-water systems
Gamma and neutron attenuation 63, 67
Neutron attenuation 226
Neutron capture gamma rays in 185, 186
Shielding properties 210

Lead
Gamma attenuation 11, 12, 38, 62, 115, 175A, 236, 297
Gamma buildup factors. 252
Gamma dose-rate buildup factors 39
Gamma energy and angular distributions through 143
Gamma penetration 290, 330
Gamma scattering 206, 328
Gamma shielding properties 99, 223, 224
Gamma transmission 27
Manufacture of shields. 175

Lead
- Neutron attenuation 26, 44, 56
- Neutron elastic scattering. 133
- Neutron shielding properties 224
- Radiation-induced melting 75
- Shielding properties 14, 121, 220, 287
- Shielding properties for spacecraft. 20
- Shielding thickness calculations 302

Lead glass
- Gamma shielding properties. 223
- Shielding properties 121

Lead-polyethylene systems
- Gamma scattering 206

Lead-water systems
- Gamma and neutron distribution 68
- Neutron attenuation 26

Lead-wax systems
- Neutron shielding properties 242

Limestone
- Gamma shielding properties 128

Lithium
- Neutron shielding properties 96

Lithium fluorides
- Neutron shielding properties 96

Lithium hydrides
- Neutron attenuation 114

Materials
- Gamma absorption coefficients 248
- Gamma activation dose rates 84
- Gamma transmission through slabs 254
- Shielding properties 71

Naval vessels
- Gamma penetration 306
- Shielding 58, 181, 217
- Shielding calculations 183, 184
- Shielding factors 124

Neutron probes
Shield design 277

Neutron radiation
Attenuation by concrete. 47

Neutrons
Absorption cross sections 231
Absorption by boron and carbon 221
Absorption coefficients in concrete 209
Absorption in fails 117
Absorption and scattering in shielding 269
Attenuation by boral 24
Attenuation in concrete. 25, 34, 246
Water-steel systems 264
Attenuation in iron and lead. 44
Attenuation by water-lead mixtures 26
Attenuation in lithium hydrides 114
Attenuation in shielding materials 17, 18, 45, 63, 144, 297
Capture in shields 17, 18
Energy spectra in air 324
Diffusion
Theory for shielding 2
Distributions in water shielding 8
Fluxes in Bulk Shielding Reactor 33
Fluxes in shields, calculations 7
Penetration, Monte Carlo method 41
Penetration in B and Pb loaded polyethylene 41
Penetration in shields 28, 113
Removal cross section, calculation 125, 126
Scattering in air. 325, 326
Scattering from materials 133
Spectral measurements 104
Transmission through boral 3, 4
Transmission through borated polyethylene 5
Transport in air 77
Transport through iron slabs 50

Oak
Gamma shielding properties 128

Oil
Shielding properties 287

Paints
Protection against nuclear thermal radiation 228, 274

Paraffin
Shielding properties 287

Personnel
Radiation recovery 53
Radiation shielding 42

Plastics
Shielding properties 147, 287

Plexiglas
Gamma shielding properties 43, 223

Polyethylene
Gamma attenuation 236
Gamma and neutron shielding properties 224
Neutron penetration in Pb and B loaded 41
Proton attenuation 235

Propylene polymers
Shielding properties 14

Protons
Shielding materials 6, 159, 160, 161
Shielding of space vehicles against 195

Radiation
Disordering of solids by corpuscular 279

Radiation protection 40

Reactor shielding
Literature survey 37

Safety glass
Gamma shielding properties 223

Sand
Gamma shielding properties 128

Sandstone
Gamma shielding properties 128

Silt
Gamma shielding properties 128

Silver
Neutron elastic scattering 133

Shelters
Design for lunar exploration 262A
Gamma shielding 98

Shelters (cont'd)
Radiation streaming in entranceways 298
Shielding effectiveness of underground 276
Shielding volume for underground 190

Shielding
Absorption and scattering of radiation by pebble 233
Aircraft nuclear 122, 123
Angular and total flux distribution calculation 87
Applications to shelters 119
Application of super conducting coils 193
Bibliography 237, 255, 312
Calculations, procedures for 73
Calculations for reactors 194
Computation of ship 124
Computer code for axially symmetric 157
Computer codes for neutron transmission 288
Computer program 199, 200
Cooling of thermal reactor 244
Design. 76, 120, 132, 139, 140, 146, 210, 286, 297, 315, 316, 338
Design of aircraft 243, 291, 292
Design and materials 257
Design of neutron 197
Design for space vehicles 159, 160, 161, 262, 270
Development 107
Development of high temperature 145
Dose buildup factors 85
Gamma attenuation 32, 113, 180, 251, 319
Gamma buildup factors 166
Gamma buildup in multi-layered 281
Gamma heating calculations 102
Gamma penetration paint source program 89
Gamma scattering 206
Manned space systems 261
Manufacture of lead 175
Material development 273, 327
Materials for space flight 162
Multilayered for space vehicles. 91
Naval requirements 168
Neutron absorption and scattering 269
Neutron angular and total flux distribution 88
Neutron attenuation 32, 113, 144, 251,278, 308
Modeling technique for determining 23
Optimization of aircraft 214
Partial for aircraft servicing 207
Partial-body 182, 189
Penetration by cosmic radiation 112
Principles of radiation 253
Properties of concretes as 179

Shielding (cont'd)
Radiation attenuation by laminated 203
Radiation penetration through ducts 141
Radiological requirements of Army 218
Research program 205
Research and theory 280
Radiation effects 169
Shape and weight 167
Technique outline 155
Technology 275
Thickness, method of calculating 108

Shielding materials
Radiation effects 130

Shielding windows
Light transmittance through 86
Optical properties of irradiation facility 72

Soil
Gamma attenuation 320
Gamma penetration 59
Gamma shielding properties 128
Neutron attenuation 282
Shielding properties 284

Solar radiation
Shield requirements 235, 263

Space
Radiation hazards of exploration 95

Space flight
Radiation hazards 304
Shielding requirements for 500 km orbit 20
Solar proton effects on manned 9
Split scatter shield design 30

Space vehicles
Shield weight calculations 322
Shielding 22, 91, 92, 159, 160, 161
Shielding for cosmic radiation 70, 112, 195, 258A, 262
Shielding for manned 6, 21, 106, 261
Shielding for solar radiation 216, 232
Shielding design 170, 171, 193, 270
Shielding materials 162
Shielding requirements 138

Space vehicles (cont'd)
Space environment effects 127

Steel
Gamma attenuation 43
Gamma shielding properties 223

Steel-water systems
Gamma and neutron distribution 68

Structures
Radiation shielding 90

Systems for nuclear auxiliary power
Shield design 111, 165

Tanks
Radiation shielding research 272

Tar
Shielding properties 287

Trucks
Shielding development 129

Tungsten
Shielding properties 14

Tungsten-uranium alloys
Shielding properties 204

Uranium
Gamma shielding properties 99

Vinyl chloride polymers
Shielding properties 81

Water
Gamma buildup factors 252
Gamma attenuation 12, 63, 125A, 297
Gamma distribution 64, 158
Gamma dose rate buildup factors 39
Neutron attenuation 56, 63
Neutron distribution 8, 64
Radiation dose rates and thermal fluxes 54
Shielding properties 121, 128, 287, 284
Shielding thickness calculations 302

Zinc bromides
    Shielding properties 52

Zirconium
    Neutron cross sections 213